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GAMMA HEATING IN LMFBR MEDIA

by

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ABSTRACT

State-of-the art approaches for the calculation of gamma heating in the core, blanket and reflector regions of LMFBR's have been evaluated, with particular emphasis on coupled neutron-gamma methods/cross section sets. The major source of calculational error was found to be the apparent failure to impose a mass-energy balance on total gamma energy yield from neutron capture and other interactions in the preparation of representative neutron-gamma cross section sets.

The applicability of many simplifying assumptions was demonstrated, including: volume-weighted homogenization, insensitivity to the shape of the gamma-source-spectrum, gamma energy deposition equal to gamma energy source more than ~ 10 cm inside large zones of uniform composition, and the negligible effect of bremsstrahlung.

A simple one-group method was developed to permit rapid, accurate estimation of the large (factor of 2) changes in the gamma energy deposition-to-source ratio possible near region interfaces. The approach, which also ensures conservation of mass-energy, was used in conjunction with coupled neutron-gamma computations to verify that previous experimental measurements of gamma heating in an LMFBR blanket mockup at M.I.T. were in accord with theoretical expectations within the experimental precision of $\sim \pm 10\%$.

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Chapter 1

INTRODUCTION

1.1 Foreword

Fast breeder reactors appear destined to play a major role in meeting the growing demand for economical and clean energy for the years ahead. The Liquid Metal (sodium) Cooled Fast Breeder Reactor (LMFBR) is the subject of concentrated efforts on an international basis and the Gas (helium) Cooled Fast Breeder Reactor (GCFR) is being pursued both in the U.S. and Europe. Since fast breeder reactors inherently require high core power densities and the presence of a blanket surrounding the core, a number of unique neutronic and photonic problems arise in their analysis. One of these special problems--that of gamma heating--is the subject of the present work.

In any reactor, and especially in the fast breeder, establishment of the thermal energy source distribution is an essential task, as it determines the temperature field within the reactor and thus the heat transport, thermal stresses and many other temperature-dependent physical and chemical properties of reactor materials. The thermal energy source in a nuclear reactor can be ultimately associated with the slowing down of heavy charged particles (fission fragments), alpha and beta particles, electrons and recoil nuclei and atoms, which are, in turn, produced by the interaction of neutrons

and photons with the reactor materials. It is convenient to distinguish between charged and uncharged particles, because in the case of charged particles, the assumption of local deposition of their kinetic energy as thermal energy is generally acceptable (A-1). On the other hand the mean free path of photons and neutrons is of the order of several centimeters, necessitating in general a detailed transport calculation.

It should be noted, however, that part of the kinetic energy of charged particles as they decelerate in the nuclear electric field can be re-emitted in the form of bremsstrahlung radiation; especially in the case of energetic beta particles and electrons. To the extent that this process is of importance in a given medium, the assumption of local deposition of the total kinetic energy of these charged particles will not be valid, because the secondary photons thus created can travel significantly far from the point of origin in the medium.

Investigation of the production of photons and their subsequent interaction with reactor materials, finally leading to energy deposition, usually referred to as Gamma Heating, is a current priority problem area in FBR design activity. Specification of gamma heating is especially important in the blanket region of the reactor, where a significant period of time is required before

bred plutonium generates a substantial amount of fission energy. During this period gammas are the dominant source of heating, particularly in the outer rows of the radial blanket. Gammas are always the primary source of heating in the reflector external to the blanket.

The purpose of this report is to investigate the available calculational tools for gamma heating analysis in FBR media. Specific objectives of this work have been:

1. To evaluate the effects of heterogeneity on gamma production and energy deposition in FBR lattices.
2. To evaluate the relatively sophisticated models presently available for gamma heating calculations (specifically the coupled neutron-gamma cross section sets), for gamma heating analyses in FBR media, and to suggest simplifications that can be adopted.

While in the interests of specificity we will employ LMFBF-oriented examples in this report, most of the work is directly applicable to the GCFR as well.

1.2 Background

Basically, gamma heating analysis for a nuclear reactor involves four steps:

1. In the first step, the neutron flux distribution is calculated utilizing neutron interaction cross sections in an appropriate diffusion or transport code.

2. Next, the gamma source distribution induced by the neutron flux is calculated. This energy-dependent gamma source distribution is obtained at each spatial point from the neutron fluxes utilizing gamma production cross sections.

3. In the third step, the gamma flux distribution is determined by performing a fixed source gamma transport calculation using gamma interaction cross sections.

4. The gamma heating rate is then determined by multiplying the gamma fluxes by the gamma energy absorption cross sections.

This step-by-step procedure was used, for example, by Stenstrom, in 1972, for gamma heating calculations for a uranium-blanketed EBR-II-type core (S-1). The neutronic calculations were done with 29 energy groups in the S_4P_3 approximation and the gamma transport calculations used 20 energy groups and the S_8P_3 approximation.

In recent years, however, a coupled neutron-gamma calculation technique that combines the above steps into a single transport calculation has been widely used. The technique uses a code originally written for single-particle transport. (ANISN(E-1), DOT(M-1), MORSE(S-2) and several others) to carry out two-particle transport calculations by introducing a special arrangement of the neutron cross sections, gamma-yields and gamma cross sections. These cross sections are commonly designated as

Coupled Neutron-Gamma Cross Section Sets, and their use is frequently deemed to be more convenient than running successive neutronic and photonic calculations. Two typical examples of particular interest, because they were used in the present study, are:

1. A 40-group coupled neutron-gamma P_3 cross section library in ANISN format (22 groups of neutrons and 18 groups of gammas) compiled at Oak Ridge National Laboratory (ORNL) in 1972 for shielding analysis of PWR depleted uranium shipping casks (M-2). At M.I.T., Wood (W-1), Brown (B-1) and Scheinert (S-3) have previously used this coupled cross section library for gamma heating analysis of LMFBR blankets and blanket assemblies irradiated in the Blanket Test Facility (BTF) at the M.I.T. Reactor. An updated version of this cross section library has become available since then (O-1).

2. A 42-group coupled neutron-gamma cross section library, with 30 groups of neutrons and 12 groups of gammas, generated by LASL/TD Division for use in vulnerability and weapon's effect investigations (S-4).

Although much more will be said on this point later (Chapter 3 in particular), it is appropriate to indicate that experience obtained in using these coupled cross section libraries for gamma heating analysis in FBR media during the course of this work has indicated that at the present state of evolution of these coupled libraries

(especially with regard to the gamma production cross sections) a simpler approach, with explicit emphasis on energy conservation in the specification of neutron-induced gamma yields, is likely to be more reliable, and sufficiently accurate.

1.3 Outline of the Present Work

The remainder of this chapter discusses sources of gammas in a nuclear reactor (Section 1.4), gamma interaction processes of importance in the reactor (Section 1.5) and the general format of coupled cross section sets (Section 1.6).

Chapter 2 shows that the heterogeneity of the fuel lattice in a typical fast reactor has negligible effect on gamma heating analysis.

Chapter 3 summarizes the experience obtained in using the coupled neutron-gamma cross section sets for FBR gamma heating calculations and suggests a simple method which is deemed to be accurate enough, and probably more reliable, at the present stage of neutron-gamma cross section set development. The effect of bremsstrahlung on the spatial redistribution of gamma energy deposition is also investigated.

Chapter 4 summarizes this investigation reiterating the main conclusions, and suggesting areas that need further work.

1.4 Sources of Gammas in a Nuclear Reactor

Gamma-sources in a reactor are:

1. Nuclear fission gammas.
 - a. Prompt fission gammas.
 - b. Short-lived fission product decay gammas.
 - c. Long-lived fission product decay gammas.
2. Capture Gammas
 - a. Prompt capture gammas.
 - b. Post-capture decay gammas.
3. Inelastic scattering gammas.
4. Gammas from (n,2n) and (n, charged particle) reactions).
5. Annihilation gammas.
6. Bremsstrahlung gammas.

A brief discussion of these processes and their relative importance in a fast breeder reactor environment follows.

Prompt gammas from fission are those emitted in coincidence with the fission event. For all practical purposes, short-lived fission product decay gammas can also be considered in apparent coincidence with the fission event.

For long-lived fission product decay gammas, times involved are long enough to identify the various radioactive nuclides among the fission products and to measure their characteristic radiations. Although they constitute

significant sources of radiation after shutdown, long-lived fission product decay gammas are of little importance during reactor operation.

Radiative capture of neutrons, the (n,γ) reaction, is the next most important source of gammas after fission. As a result of the absorption of a neutron, the nucleus gains excitation energy equal to the binding energy (B) of the neutron plus the kinetic energies (E_c) of the neutron and the target nucleus in the center-of-mass system. In the radiative capture reaction, this excitation energy is lost by prompt emission of gammas. It should be noted that if the product nucleus is radioactive, or is formed in an isomeric state, its post-capture decay gamma-energy may be treated separately, or may be included in the binding energy, B , of the neutron.

A special case of interest here is the (n,γ) reaction in Na; it produces the Na^{24} isotope with a half-life of 14.8 hrs., emitting, per decay, 1.38 and 2.76 Mev gammas in cascade. Since the sodium coolant is free to move and mix, its post-capture decay gammas should be treated separately from those of the fixed structure and fuel (K-1).

It may also be pointed out that a (nearly negligible) part of the excitation energy (E_c+B) goes into the recoil of the nucleus as the gamma(s) is (are) emitted. This will, of course, be true for all gamma emission processes.

The recoil energy of the nucleus is deposited locally.

Inelastic scattering can occur if E_c is greater than ϵ_1 , the energy of the first excited state of the target nucleus. Within a very short time after the scattering (on the order of 10^{-14} sec), the nucleus loses its energy of excitation by emitting one or more gammas.

Reactions other than capture and inelastic scattering can lead to excited residual nuclei which lose their energy of excitation by photon emission. Three such reactions are (n, p) , (n, α) and $(n, 2n)$. However, thresholds for these reactions are high and cross sections low, making their contribution negligible in most situations.

Annihilation gammas are produced mainly from electron-positron pairs arising from pair production events at higher gamma energies. (Positron sources other than pair production events are rare in the reactor since neutron and fission reactions usually lead to nuclei which are on the excess neutron side of the stability line and, therefore, decay by β^- emission). Since the pair production process is a significant form of gamma energy deposition in the reactor, annihilation radiation is quite important. In coupled neutron-gamma cross section sets, this is appropriately accounted for as a gamma down-scatter event (Section 1.6).

Bremsstrahlung radiation will be considered separately in Chapter 3 (Section 3.8).

1.5 Gamma Energy Deposition Reactions

In a nuclear reactor environment, three types of gamma interactions with matter must be considered:

1. Compton scattering.
2. Pair production.
3. Photoelectric effect.

These three events cover the range of important gamma interactions (E-2). Compton scattering, more important in the range from 0.1 to 2.0 Mev, reduces the gamma energy without the extinction of the gamma itself. The energy thus lost is imparted to an electron. Pair production can occur only if the energy of the photon is greater than 1.02 Mev, which is the minimum needed to create an electron-positron pair, the excess being available as the kinetic energy of the pair and the interacting nucleus. The photoelectric effect is dominant at low energies (less than 100 kev). Here, the gamma is absorbed, transferring all its energy to an electron.

1.6 The General Format of Coupled Neutron-Gamma Cross Section Sets

Table 1.1 gives the format of the 40-group ORNL cross section set. The first column lists the discrete ordinate S_N format for cross sections used in ANISN, i.e., $\overline{E\sigma_a}$, $\overline{v\sigma_f}$, σ_t , σ_{gg} , $\sigma_{g-1 \rightarrow g}$, $\sigma_{g-2 \rightarrow g}$, ----, $\sigma_{g-39 \rightarrow g}$. The table length in the vertical direction is 43. The 22 neutron and 18 gamma group numbers are labeled along the

Table 1.1
Format of 40-Group ORNL Coupled Neutron-Gamma Cross Section Set

Cross Section	Neutron Groups						Gamma Groups			
	1	2	----	----	----	22	23	----	39	40
σ_a or $\overline{E}\sigma_a$			σ_a					$\overline{E}\sigma_a$		
$\overline{\nu}\sigma_f$							$\overline{\nu}\sigma_f$ (Gammas) = 0			
σ_t		Neutron Transport					Gamma Transport			
σ_{gg}	$\sigma_{1\rightarrow 1}$		Cross Sections				$\sigma_{23\rightarrow 23}$	Cross Sections		
$\sigma_{g-1\rightarrow g}$	0	$\sigma_{1\rightarrow 2}$					$\sigma_{22\rightarrow 23}$	---		
$\vdots$	0	0	----					---	----	
$\sigma_{g-17\rightarrow g}$	0	0	0	---					$\sigma_{22\rightarrow 39}$	$\sigma_{23\rightarrow 40}$
$\vdots$	0	0	0	0	----		Gamma Production			----
$\sigma_{g-21\rightarrow g}$	0	0	0	0	0	$\sigma_{1\rightarrow 22}$		Cross Sections		
$\sigma_{g-22\rightarrow g}$	0	0	0	0	0	0	$\sigma_{1\rightarrow 23}$			
$\vdots$	0	0	0	0	0	0	0	---		
$\sigma_{g-38\rightarrow g}$	0	0	0	0	0	0	0	0	$\sigma_{1\rightarrow 39}$	
$\sigma_{g-39\rightarrow g}$	0	0	0	0	0	0	0	0	0	$\sigma_{1\rightarrow 40}$

top. Photofission (γ, f) and photoneutron (γ, n) reactions²⁶ and energy upscatter mechanisms for neutrons or gammas are not allowed for.

Gamma production cross sections are entered as down-scattering events from 22 groups of neutrons to 18 groups of gammas. These cross sections were generated at ORNL using the POPOP4 Code (F-1), and are documented in Ref. 0-1.

Gamma scattering cross sections are expanded in a P_3 Legendre expansion using the Klein-Nishina approximation (W-2). The zeroth moment photon transfer cross sections are modified for the pair production event as follows (K-2):

$$\sigma_s^m(E \rightarrow E') = \sigma_{s0}(E \rightarrow E') + \sigma_{pp}(E) \cdot 2\delta(E' - 0.51), \quad (1.1)$$

where

$\sigma_{s0}^m(E \rightarrow E')$ is the modified zeroth moment photon transfer cross section,

$\sigma_{s0}(E \rightarrow E')$ is the zeroth moment photon transfer cross section as given by the Klein-Nishina approximation,

$\sigma_{pp}(E)$ is the pair production cross section,

E is the energy of the incident photon,

E' is the energy of the scattered photon, and

δ is the Kronecker delta.

Use of the modified zeroth moment photon transfer cross

sections in transport codes takes care of annihilation radiation subsequent to pair production events.

The total cross section for a given group is obtained from the sum of the Klein-Nishina total scattering cross section (unmodified) for that group plus the group average cross sections for the photoelectric effect and pair production.

The energy absorption cross section, $\overline{E\sigma}_a$ (Mev-barn) is defined by (K-2):

$$\begin{aligned} \overline{E\sigma}_a(E) = & (E-1.02)\sigma_{pp}(E) + E\sigma_{pe}(E) \\ & + \sum_{E'} (E-E')\sigma_{so}(E \rightarrow E'), \end{aligned} \quad (1.2)$$

where $\sigma_{pe}(E)$ is the cross section for the photoelectric effect, and all other symbols are the same as in Eq. 1.1.

MUG (K-2) and GAMLEG69, an updated version of GAMLEG (L-1), are suitable Codes for preparing the gamma transport cross sections in this format.

The format for the LASL 42-group cross section set (see Ref. S-4) is similar except that the first cross section position contains the absorption cross section (i.e. the sum of photoelectric and pair production cross sections) rather than the energy absorption cross section as given by Eq. 1.2. Since the absorption cross section per se is not used in discrete ordinate codes, the results are not affected by the quantity entered in table position 1. It should also be noted that the term $(2L+1)$ in higher

order Legendre expansions is not included in the LASL cross section set (D-1). This term is included in ORNL cross sections.

Table 1.2 lists the neutron group structure and Table 1.3 gives the gamma group structure for the two cross section sets. In Table 1.2, the neutron group structure of a 26-group Russian (also known as AB3N or Bondarenko) cross section set (B-2) is also included for later reference.

Table 1.2

Neutron Energy Group Structures of ORNL and LASL Coupled Cross Section Sets and 26-Group Russian Cross Section Set

Group g	ORNL Coupled Cross Section Set E _{max} = 15.0 Mev		LASL Coupled Cross Section Set E _{max} = 17.0 Mev		26-Group Russian Cross Section Set E _{max} = 10.5 Mev	
	Lower Energy Boundary, Group g					
1	12.2	Mev	15.0	Mev	6.5	Mev
2	10.0	Mev	13.5	Mev	4.0	Mev
3	8.18	Mev	12.0	Mev	2.5	Mev
4	6.36	Mev	10.0	Mev	1.4	Mev
5	4.96	Mev	7.79	Mev	0.3	Mev
6	4.06	Mev	6.07	Mev	0.4	Mev
7	3.01	Mev	3.68	Mev	0.2	Mev
8	2.46	Mev	2.865	Mev	0.1	Mev
9	2.35	Mev	2.232	Mev	46.5	kev
10	1.83	Mev	1.738	Mev	21.5	kev
11	1.11	Mev	1.353	Mev	10.0	kev
12	0.55	Mev	0.823	Mev	4.65	kev
13	0.111	Mev	0.5	Mev	2.15	kev
14	3.35	kev	0.303	Mev	1.0	kev
15	0.583	kev	0.184	Mev	0.465	kev
16	0.101	kev	67.6	kev	0.215	kev
17	29.0	ev	24.8	kev	0.100	kev
18	10.7	ev	9.12	kev	46.5	ev
19	3.06	ev	3.35	kev	21.5	ev
20	1.12	ev	1.235	kev	10.0	ev
21	0.414	ev	0.454	kev	4.65	ev
22	Thermal		0.167	kev	2.15	ev
23			61.4	ev	1.00	ev
24			22.6	ev	0.465	ev
25			8.32	ev	0.215	ev
26			3.06	ev	Thermal	
27			1.13	ev		
28			0.414	ev		
29			0.152	ev		
30			Thermal			

Table 1.3

Gamma Energy Group Structures of ORNL and LASL Coupled
Cross Section Sets

Group g	ORNL Coupled Cross Section Set E _{max} = 10 Mev Lower Energy Boundary, Mev	Group g	LASL Coupled Cross Section Set E _{max} = 10 Mev Lower Energy Boundary, Mev
23	8.0	31	9.0
24	6.5	32	8.0
25	5.0	33	7.0
26	4.0	34	6.0
27	3.0	35	5.0
28	2.5	36	4.0
29	2.0	37	3.0
30	1.66	38	2.0
31	1.33	39	1.0
32	1.00	40	0.5
33	0.80	41	0.1
34	0.60	42	0.01
35	0.40		
36	0.30		
37	0.20		
38	0.10		
39	0.05		
40	0.01		

Chapter 2

EFFECT OF UNIT CELL HETEROGENEITY ON FBR GAMMA HEATING ANALYSIS

2.1 Introduction

In the course of neutronic calculations for fast breeder reactors, heterogeneous effects are usually considered as a posteriori corrections to a volume-averaged homogeneous representation as in Eqs. 2.1 and 2.2 below:

$$N_i = \sum_j N_{ij} V_j / \sum_j V_j, \quad \text{and} \quad (2.1)$$

$$\Sigma_x^g = \sum_i N_i f_{x,i}^g \sigma_{x,i}^g, \quad (2.2)$$

where

g is the group number,

x is the process (e.g., absorption),

j is the region (e.g., fuel region),

i is the material (e.g., U-238),

N is the number density,

V is the volume,

σ is the infinite dilution value of the microscopic cross section,

f is the resonance energy-self-shielding factor calculated for the homogeneous medium, and

Σ is the macroscopic cross section.

For gamma heating analysis, the following simplifications can be noted:

1. In the absence of a photonic chain reaction, how a given generation of photons is lost, whether by absorption in material 1 or 2, or by absorption in group i or j , or by leakage, does not affect the next generation of photons. Because of this, second order perturbations or local effects in photonic reactions do not acquire any global importance for the entire reactor. For example, the leakage of gammas from a bare core will not affect the gamma heating analysis inside the core (Section 2.4.1). On the other hand the leakage of neutrons will have to be investigated in more detail because of its effect on the multiplication constant of the reactor. Calculations reported in Section 2.4, where leakage calculations are done in both homogeneous and heterogeneous geometries, prove this point.

It may also be noted that the mean free path of gammas in FBR media is somewhat less than that of neutrons (Section 2.2), making leakage less important, but raising questions with respect to the possible effects of spatial self-shielding on gamma heating calculations. The phenomenon of anisotropic diffusion of photons, unimportant for its effect on axial leakage (because leakage itself has no global effect on gamma heating analysis),

could also have some effect on the spatial flux ratios along the axis of the unit cell. Here too, as will be shown in Section 2.3, the basic effect is itself negligible; hence perturbations are also inconsequential.

2. Gamma interaction cross sections do not exhibit the sharp resonances characteristic of neutron cross sections (see Figs. 2.1 and 2.2). Since energy self-shielding factors arise almost entirely because of resonance peaks, the energy self-shielding factors for gamma transport can be taken as unity. This applies to both homogeneous and heterogeneous cases. Gamma cross sections, however, do show structure (the K-edge) at low energies. This effect will be analyzed in Section 2.5.

The remainder of this chapter follows the course indicated in the above discussion. Section 2.6 summarizes the conclusions.

2.2 Gamma Mean Free Paths in LMFBF Media

The mean free path, λ , for a particular interaction is given by:

$$\lambda = 1/\Sigma, \quad (2.3)$$

where Σ is the macroscopic interaction cross section.

The macroscopic transport cross section for gammas can be written as:

Fig. 2.1 Gamma Energy Absorption Cross Sections for Uranium, Iron and Sodium

Fig. 2.2 Microscopic Cross Sections for Photoelectric, Pair Production and Compton Scattering Interactions in Uranium.

$$\begin{aligned}
 \Sigma_{tr}^g &= \Sigma_t^g - \mu_g^g \Sigma_s^g \\
 &= \Sigma_a^g + (1 - \mu_g) \Sigma_s^g,
 \end{aligned}
 \tag{2.4}$$

where

Σ_t^g is the total cross section in group g ,

Σ_a^g is the absorption cross section in group g

(i.e., the sum of pair production and photo-electric cross sections),

Σ_s^g is the Compton scattering cross section in group g , and

μ_g is the average cosine of the Compton scattering angle in group g .

μ_g is defined by Eqs. 2.5 and 2.6 (D-2):

$$\mu_g = \frac{\int_{-1}^1 \mu k(\mu) d\mu}{\int_{-1}^1 k(\mu) d\mu}, \tag{2.5}$$

$$k(\mu) = \frac{1 + \mu^2}{[1 + \alpha_g(1 - \mu)]^2} + \frac{\alpha_g^2(1 - \mu)^2}{[1 + \alpha_g(1 - \mu)]^3}, \tag{2.6}$$

where

$$\alpha_g = E_g / 0.511, \text{ and}$$

E_g is the average gamma energy (Mev) in group g .

Table 2.1 shows the values of μ in the 12 gamma energy group structure, obtained by numerical integration of Eq. 2.5.

Table 2.1

Average Cosines of Compton Scattering Angles for the 12
Gamma-Energy-Group Structure

Group(g)	Gamma Energy Range (Mev)	Average Cosine, μ_g
1(31)*	10.0 - 9.0	0.5560
2(32)	9.0 - 8.0	0.5478
3(33)	8.0 - 7.0	0.5382
4(34)	7.0 - 6.0	0.5270
5(35)	6.0 - 5.0	0.5135
6(36)	5.0 - 4.0	0.4970
7(37)	4.0 - 3.0	0.4756
8(38)	3.0 - 2.0	0.4460
9(39)	2.0 - 1.0	0.3993
10(40)	1.0 - 0.5	0.3319
11(41)	0.5 - 0.1	0.2326
12(42)	0.1 - 0.01	0.0738

*Numbers in parentheses refer to gamma group numbers
in the LASL coupled cross section set.

One-group gamma cross sections are then obtained by averaging over an appropriate gamma spectrum.

$$\Sigma_x = \sum_g \Sigma_x^g \phi_g / \sum_g \phi_g, \quad (2.7)$$

where x denotes the process (e.g., transport) and ϕ_g is the total flux (photons/cm²-sec) in group g . Three gamma spectra appropriate to the core, blanket and steel reflector of the reference LMFBR (discussed in Chapter 3), shown in Fig. 2.3, were used to obtain the one-group values. Table 2.2 shows the mean free paths for gamma absorption, scattering, transport and total interactions, as obtained from Eq. 2.3. Approximate values of absorption and transport mean free paths for neutrons are also given for comparison. These values were obtained using a one-group LMFBR cross section set from Ref. W-3.

Fig. 2.3 Gamma Spectra in Reference LMFBR's Media

Table 2.2

Gamma Mean Free Paths in Reference LMFBF Media (cm)

Mean Free Path	Core	Blanket	Reflector (steel)
Absorption	3.44 (200) *	2.60 (200)	7.64 (1000)
Scattering	4.05	3.04	2.07
Transport	2.21 (5.5)	1.68 (4.0)	2.10 (3.5)
Total	1.85	1.40	1.60

* Numbers in parentheses are approximate mean free paths for neutrons, given for comparison.

2.3 Gamma Heating Analysis for the Unit Cell

Figure 2.4 shows a representative cylindricized three-region unit cell for an LMFBR core zone. Region 1 is the fuel region; region 2 is the clad region; and region 3 is the coolant region, whose outer radius is defined by assigning to each fuel rod its share of the coolant. This cylindricalization of the unit cell is identical to that used by Gregory (G-2), and a white boundary condition, one which returns all neutrons and gammas from the boundary with an isotropic distribution, is used.

Tables 2.3 and 2.4 compare the results of gamma heating calculations for this three-region unit cell and a corresponding one-region volume-homogenized unit cell. Calculations were done using the S_8P_3 option in ANISN. These two tables, together with the introductory comments in Section 2.1, essentially sum up the effects related to the heterogeneity of the unit cell in FBR gamma heating calculations, and thus set the tone for the remainder of this chapter. The following points are worthy of note:

1. There is a difference of about 50% in gamma energy yields between Tables 2.3 and 2.4. This is representative of the difference in gamma heating values obtained using the ORNL and the LASL cross section sets and is discussed in detail in Chapter 3.

Region Parameters:

$$r_1 = 0.3366 \text{ cm}$$

$$r_2 = 0.3810 \text{ cm}$$

$$r_3 = 0.4841 \text{ cm}$$

Isotopic number densities (units of $10^{24}/\text{cm}^3$):

Fuel:	U-238	0.0182
	Pu-239	0.002435
	0	0.04127
Clad:	Fe	0.0848
Coolant:	Na	0.0224

Fig. 2.4 A Representative Three-Region Unit Cell for an LMFBR Core

Table 2.3

Comparison of Gamma Heating Calculations for Heterogeneous and the Volume-Homogenized Unit Cells Using the ORNL Cross Section Set

Quantity of Interest Mev/cm-sec*	Heterogeneous Unit Cell	Volume-Homogen- ized Unit Cell
Gamma Energy Produced Total	9.834 [†]	9.833
From U-238	4.273	4.270
From Pu-239	5.053	5.056
From Fe	0.377	0.377
From Na	0.127	0.127
From O	0.003	0.003
Gamma Energy Absorbed Total	9.757	9.745
In U-238	7.024	7.014
In Pu-239	0.996	0.995
In Fe	0.965	0.966
In Na	0.286	0.287
In O	0.486	0.485

* Normalized to a linear Power density of 1 fission neutron/cm-sec

[†] The number of figures retained are only for comparison between the two cases and are not meant to suggest the accuracy of the gamma yield data (compare with Table 2.4; see text and Chapter 3).

Table 2.4

Comparison of Gamma Heating Calculations for Heterogeneous
and the Volume-Homogenized Unit Cells Using the LASL Cross
Section Set

Quantity of Interest Mev/cm-sec*	Heterogeneous Unit Cell	Volume-Homogenized Unit Cell
Gamma Energy Produced		
Total	6.303 [†]	6.326
From U-238	3.041	3.039
From Pu-239	2.662	2.689
From Fe	0.518	0.517
From Na	0.080	0.078
From O	0.003	0.003
Gamma Energy Absorbed		
Total	6.393	6.417
In U-238	4.599	4.610
In Pu-239	0.647	0.648
In Fe	0.646	0.653
In Na	0.186	0.188
In O	0.316	0.317

* Normalized to a linear power density of 1 fission neutron/
cm-sec

[†] The number of figures retained are only for comparison
between the two cases and are not meant to suggest the
accuracy of the gamma yield data (compare with Table 2.3;
see text and Chapter 3).

2. The values in Tables 2.3 and 2.4 are normalized to a unit fission neutron production rate in one cm axial length of the unit cell, i.e.,

$$\sum_g \int_0^{r_3} v \Sigma_f^g(r) \cdot \phi_g(r) \cdot 2\pi r dr = 1, \quad (2.8)$$

or, in other words, to a linear power density of 1 fission neutron/cm-sec. Note that ANISN will equate the above expression to the eigenvalue (k) if a normalization factor of unity is specified in the code input. It may be pointed out that this eigenvalue remains unaffected by the presence of the gamma groups, because all fission cross section entries in gamma groups are zero (see Table 1.1).

3. In Tables 2.3 and 2.4, there is some discrepancy (of the order of 1%) between the total gamma energy produced and absorbed although the white boundary condition with an albedo of unity was used. This discrepancy was also noted in infinite medium calculations done in planar geometry, with mirror boundary conditions. It is apparently due to slight discrepancies or round-off errors in the gamma-energy absorption cross sections, and could be remedied by slight revision of these cross sections to restore consistency. However, in view of much larger errors introduced elsewhere, referenced

throughout the present work, the energy absorption cross sections supplied as part of the cross section sets have been used. This discrepancy should not be confused with the effect of heterogeneity. Comparison of corresponding values along horizontal lines in Tables 2.3 and 2.4 correctly assesses the heterogeneous effects.

4. In the gamma heating calculations for the heterogeneous and the volume-homogenized unit cells, no attempt was made to keep all gamma-producing neutronic reactions identical by compensating for the spatial depression of the neutron flux in the heterogeneous case. However, this is partly compensated, in an integral sense, through the normalization of Eq. 2.8 (note that the multiplication constant k is slightly different in the two cases. The heterogeneous reactivity contribution is in the range of 50¢ to \$1; by equating the expression in Eq. 2.8 to unity for both cases, this difference is compensated for in an integral sense). Comparison of corresponding values for the homogeneous and the heterogeneous calculations in Tables 2.3 and 2.4 shows that this is satisfactory and that more detailed compensation for neutronic heterogeneous effects is not required for the purpose of gamma heating calculations. The maximum difference between heterogeneous and homogeneous calculations of gamma heating is found to be in the case of the clad and the sodium coolant, about 1 part in 100.

Since a total of about 1 Mev of gamma energy per fission neutron is deposited in the clad and the coolant regions, this difference amounts to an additional 0.01 Mev of energy showing up in the fuel region, if heterogeneous calculations are done. This should be compared with (roughly) a total of 65 to 70 Mev deposited in the fuel region per fission neutron. Hence its importance for engineering purposes appears to be negligible, and unit cell heterogeneity is not important for FBR gamma heating calculations.

While we will not dwell on this point further here, it should be noted that this is in large part due to the fact that the fuel is at the same time the dominant source of, and sink for, gammas; over 90% of gamma energy originates, and approximately 80% is deposited, in the fuel in fast reactor cores and blankets. Kadiroglu, in work on unit cell neutronics which is also applicable to photon transport has shown this in a particularly clear fashion using a simple two-region model of the unit cell (K-3).

2.4 Effect of Heterogeneity on Gamma Leakage

2.4.1 Leakage from a Bare LMFBR Core.

Table 2.5 shows the average gamma flux ratios for the three-region unit cell of Fig. 2.4. The flux ratios were calculated using S_8P_3 option in ANISN and the ORNL

Table 2.5

Ratios of Average Gamma Fluxes for a Representative Unit Cell

Gamma Group	$\bar{\phi}_{\text{clad}} / \bar{\phi}_{\text{fuel}}$	$\bar{\phi}_{\text{cool}} / \bar{\phi}_{\text{fuel}}$
1	1.09683	1.07538
2	1.08203	1.06475
3	1.01579	1.01417
4	0.98902	0.98935
5	0.98695	0.98776
6	0.98409	0.98523
7	0.98206	0.98345
8	0.98131	0.98202
9	0.98518	0.98625
10	0.98212	0.98253
11	0.98408	0.98430
12	0.99118	0.99125
13	1.01445	1.01946
14	1.07614	1.07474
15	1.30507	1.30133
16	1.70840	1.72425
17	1.19622	1.27036
18	0.63088	0.54036

coupled cross section set (18 gamma groups). The macroscopic gamma transport cross sections are then flux-weighted using the relation:

$$\Sigma = \frac{\Sigma_1 V_1 + \Sigma_2 V_2 (\bar{\phi}_2 / \bar{\phi}_1) + \Sigma_3 V_3 (\bar{\phi}_3 / \bar{\phi}_1)}{V_1 + V_2 (\bar{\phi}_2 / \bar{\phi}_1) + V_3 (\bar{\phi}_3 / \bar{\phi}_1)}, \quad (2.9)$$

where 1,2 and 3 refer to the fuel, clad and coolant regions, respectively, as in Fig. 2.4.

Table 2.6 compares the results of gamma heating calculations for a bare LMFBR core having the same composition as the unit cell, using the flux-weighted cross sections, with those obtained by using a simple volume-homogenized prescription. The difference in gamma heating rate at each spatial point along the radius of the core is found to be negligible.

2.4.2 Leakage from a Steel-Reflected Multi-Cell Slab Reactor

Figure 2.5 shows a multi-cell slab core with a steel reflector. A relatively larger cell-size than is encountered in FBR's is selected to accentuate the effects of heterogeneous geometry. The clad is homogenized with the coolant, however, to reduce the number of regions. The gamma dose rates in the steel reflector are then calculated using first the detailed multi-cell heterogeneous representation of the core, and then an equivalent volume-homogenized model. The S_8P_1 option in ANISN

Table 2.6

Gamma Heating Rates for a Bare LMFBF Core

Radial Dis- tance from the center of the core, cm	Gamma Energy Deposition, Mev/cm ³ -sec [*]	
	Using ORNL Cross Section Set With No Spatial Self Shielding	Using ORNL Cross Section Set with Gamma Cross Sections Weighted by Flux Ratios of Table 2.5
0	9.686 [†]	9.687
10	9.588	9.589
20	9.249	9.251
30	8.682	8.682
40	7.898	7.900
50	6.925	6.926
60	5.793	5.794
70	4.543	4.543
80	3.219	3.218
90	1.869	1.870
100	0.476	0.476

* Normalized to a power density of 1 fission neutron/cm³-sec at the core center; $\bar{\nu}=2.93$ may be used for conversion.

[†] Number of figures retained are only for comparison. The values, although representative, show considerable difference from those obtained using the LASL coupled cross section set (see Chapter 3).

Fig. 2.5 Steel-Reflected Multi-Cell Slab Core

was used (in the remainder of this report the S_8P_1 approximation is used in all transport calculations; see Ref. S-3 where Scheinert has shown that this is satisfactory). Table 2.7 compares the results of these calculations. Since there is no significant difference in the calculated dose rates, a volume-homogenized representation appears to be satisfactory.

2.5 Effect of Energy Self-Shielding in Gamma Heating Calculations

As pointed out earlier in Section 2.1, gamma interaction cross sections do not exhibit sharp resonances characteristic of neutron cross sections. However, as the photon energy approaches the binding energy of the K-shell electrons, the photoelectric absorption cross section exhibits a sharp peak, and "energy self-shielding" might, therefore, be expected. To assess the potential effect on gamma heating calculations, an arbitrary reduction by a factor of ten was applied to the absorption cross section in the three lowest gamma energy groups in the ORNL cross section set (gamma groups 16,17 and 18; the absorption cross section in these groups is entirely photoelectric). Gamma heating calculations were then done for the slab core of Fig. 2.5 in detailed multi-cell heterogeneous geometry.

Table 2.7

Gamma Dose Rates in a Steel Reflector Driven by a Multi-Cell Slab Core

Distance from Core-Reflector Interface, cm	Gamma Dose Rate in the Reflector, Mev/Cm ³ -sec*	
	Reflector Driven by the Multi-Cell Slab Core of Fig. 2.5 in Heterogeneous Geometry	Reflector Driven by the Volume Homogenized Slab Core
2	0.8778 [†]	0.8827
4	0.6454	0.6463
6	0.5081	0.5083
8	0.4153	0.4153
10	0.3601	0.3601
12	0.3033	0.3033
14	0.2431	0.2431
16	0.1850	0.1851
18	0.1281	0.1282
20	0.0703	0.0704

* Normalized to a linear power density of 1 fission neutron/cm-sec in cell 1 (see Fig. 2.5). For distribution of linear power in different cells, see column 1, Table 2.8.

[†] NOTE: Number of figures retained are only for comparison. These values should not be considered as representative of gamma dose rates in an LMFBR steel reflector, because for simplicity, no blanket was used between the core and the reflector.

Table 2.8 summarizes the results of these calculations. It is seen that heavy self-shielding of the lower gamma energy groups slightly increases energy deposition in the fuel region. This is primarily because self-shielding has the effect of flattening the scattering-source in these groups, increasing the relative importance of this source in the coolant region. However, even with the cell dimensions larger than those encountered in practice, and heavy self-shielding ($f = 0.1$) in the last three groups, the effect is small. The additional energy that appears in the fuel region, even in this extreme case, is of the order of 0.01 Mev/fission neutron (see, for example, fuel region, cell 4 in Table 2.8). Its importance for engineering purposes appears to be negligible.

2.6 Conclusions

Gamma heating calculations have been done for a three region LMFBR core-zone unit cell, and a steel-reflected multi-cell slab core, in heterogeneous geometries, and in equivalent volume-homogenized representations, using the discrete ordinate transport code ANISN and multigroup coupled neutron-gamma cross sections. Calculations have also been done for a bare LMFBR core

Table 2.8
Gamma Heating Rates for the Steel Reflected Multi-Cell
Slab Reactor of Fig. 2.5

Region	Linear Power Density, Fission-neut/ cm-sec	Gamma Heating Rate, Mev/cm-sec	
		Case 1: No Energy Self-Shielding	Case 2: With Energy Self-Shielding in Gamma Groups 16,17,18
Fuel, cell 1	1.0	8.502	8.512
Coolant,* cells 1-2	-	1.236	1.226
Fuel, cell 2	0.9927	8.441	8.454
Coolant, cells 2-3	-	1.218	1.209
Fuel, cell 3	0.9709	8.255	8.267
Coolant, cells 3-4	-	1.183	1.173
Fuel, cell 4	0.9352	7.956	7.964
Coolant, cells 4-5	-	1.131	1.121
Fuel, cell 5	0.8860	7.542	7.553
Coolant, cells 5-6	-	1.063	1.054
Fuel, Cell 6	0.8243	7.022	7.030
Coolant, cells 6-7	-	0.980	0.973
Fuel, cell 7	0.7517	6.407	6.414
Coolant, cells 7-8	-	0.885	0.878
Fuel, cell 8	0.6714	5.699	5.705
Coolant, cell 8-9	-	0.771	0.765 (continued)

Table 2.8 (Continued)
Gamma Heating Rates for the Steel Reflected Multi-Cell Slab Reactor of Fig. 2.5

Region	Linear Power Density, Fission-neut/cm-sec	Gamma Heating Rate, Mev/cm-sec	
		Case 1: No Energy Self-Shielding	Case 2: With Energy Self-Shielding in Gamma Groups 16,17,18
Fuel, cell 9	0.6041	4.795	4.808
Coolant, cell 9	-	0.547	0.544
Reflector, distance from interface, cm		Mev/cm ³ -sec	
2	-	0.8778 [†]	0.8731
4	-	0.6454	0.6452
6	-	0.5081	0.5087
8	-	0.4153	0.4159
10	-	0.3601	0.3604
12	-	0.3033	0.3034
14	-	0.2431	0.2432
16	-	0.1850	0.1851
18	-	0.1281	0.1281
20	-	0.0703	0.0704

* Clad homogenized with coolant.

[†] See Note Table 2.7

using gamma interaction cross sections weighted by the spatial flux ratios of the unit cell, and for the multi-cell slab core using arbitrarily-heavy energy self-shielding of the photoelectric absorption cross sections at low gamma energies. The overall conclusions that can be drawn from these calculations are, that the heterogeneity of the LMFBR fuel lattice and any energy self-shielding in gamma interaction cross sections, have negligible effect on gamma energy deposition in different materials or different regions of the lattice, and on gamma leakage within a zone, or from one zone to the other. Hence simple volume-weighted homogenization is adequate for engineering calculations of gamma heating in fast reactor cores, blankets and reflectors.

Chapter 3

GAMMA HEATING ANALYSES

In this chapter we will consider a sequence of topics bearing on the amount and quality of detail which must go into gamma heating analyses in homogenized fast reactor media. Inconsistencies in state-of-the-art formulations will be brought out which have been masked by a certain degree of over-sophistication, and some simple approaches which satisfy gamma energy balance requirements will be introduced

3.1 Introduction

A multigroup coupled neutron-gamma cross section set has essentially all the information needed to do neutron-induced gamma heating calculations. Table 3.1 shows a 4-group coupled cross section table, used only for illustrative purposes (since 4 groups is the minimum number capable of illustrating several points of interest). The gamma production cross sections, entered as down-scattering events from neutron groups 1 and 2 to gamma groups 3 and 4, include the production cross sections for all gamma-producing neutron reactions, i.e.,

$$\sigma_{13} = \sigma_{13}^c + \sigma_{13}^f + \sigma_{13}^{in} + \sigma_{13}^{n,2n} + \dots \text{etc.}, \quad (3.1)$$

$$\sigma_{14} = \sigma_{14}^c + \sigma_{14}^f + \sigma_{14}^{in} + \sigma_{14}^{n,2n} + \dots \text{etc.}, \quad (3.2)$$

Table 3.1
A 4-Group Coupled Neutron-Gamma Cross Section Table

Cross Section	Neutron Groups		Gamma Groups	
	1	2	3	4
Absorption	σ_{a1}	σ_{a2}	σ_{a3}	σ_{a4}
Fission	$\nu\sigma_{f1}$	$\nu\sigma_{f2}$	0	0
Total	σ_{t1}	σ_{t2}	σ_{t3}	σ_{t4}
Scattering:	σ_{11}	σ_{22}	σ_{33}	σ_{44}
	--	σ_{12}	σ_{23}	σ_{34}
	--	--	σ_{13}	σ_{24}
	--	--	--	σ_{14}

$$\sigma_{23} = \sigma_{23}^c + \sigma_{23}^f + \sigma_{23}^{n,2n} + \dots \text{ etc.}, \quad (3.3)$$

and

$$\sigma_{24} = \sigma_{24}^c + \sigma_{24}^f + \sigma_{24}^{n,2n} + \dots \text{ etc.}, \quad (3.4)$$

where

σ denotes the microscopic cross section, barn,

c denotes capture,

f denotes fission,

in denotes inelastic scattering, and

n,2n denotes (n,2n) reaction.

The corresponding neutron reaction cross sections can be written as:

$$\sigma_{a1} = \sigma_{c1} + \sigma_{f1} + \sigma_1^{n,2n} + \dots \text{ etc.}, \quad (3.5)$$

$$\sigma_{a2} = \sigma_{c2} + \sigma_{f2} + \sigma_2^{n,2n} + \dots \text{ etc.}, \quad (3.6)$$

and

$$\sigma_{12} = \sigma_{12}^e + \sigma_{12}^{in}, \quad (3.7)$$

where e denotes elastic scattering and other symbols are as in the preceding equations.

The following quantities are of particular interest in gamma heating calculations:

1. Neutron-Energy-Dependent Gamma Production Spectra, given by:

$$\frac{\sigma_{13}}{\sigma_{13} + \sigma_{14}}, \quad \frac{\sigma_{14}}{\sigma_{13} + \sigma_{14}}; \quad \text{Gamma Production Spectrum from Neutron Group 1,}$$

and

$$\frac{\sigma_{23}}{\sigma_{23} + \sigma_{24}}, \quad \frac{\sigma_{24}}{\sigma_{23} + \sigma_{24}}; \quad \text{Gamma Production Spectrum from Neutron Group 2.}$$

The gamma production spectra will be different for different gamma-producing neutron reactions.

There is considerable uncertainty in gamma production spectra--as large as, or even greater than, 50%. Figure 3.1 shows normalized gamma-source spectra for thermal neutron capture in U-238, taken from two different sources, Refs. B-3 and T-1, in 20-gamma-group structure. Figure 3.2 plots the same data in 5-gamma-group structure. It can be seen that even in a coarse gamma-group structure (5 groups), which should average-out some of the discrepancies, the uncertainty in a particular group can be as much as 100%. Further, data on neutron-energy-dependent gamma production spectra from all reactions and for all materials of interest is practically non-existent. It will be shown in Section 3.3 that these uncertainties in gamma production spectra are mitigated to a large extent in gamma heating calculations, if the total amount of gamma energy produced in any event is accurately preserved.

Fig. 3.1 Normalized Gamma-Source Spectrum from Thermal Neutron Capture in U-238 (20 Gamma Group)

Fig. 3.2 Normalized Gamma Source Spectrum from Thermal Neutron Capture in U-238
(5 Gamma Groups)

2. Neutron-Energy-Dependent Gamma Energy Production

Cross Sections, Mev-barn, given by:

$E_3\sigma_{13} + E_4\sigma_{14}$; Gamma Energy Production Cross Section for Neutron Group 1,

and

$E_3\sigma_{23} + E_4\sigma_{24}$; Gamma Energy Production Cross Section for Neutron Group 2,

where E_3 and E_4 are the average gamma energies in groups 3 and 4, respectively.

(Confusion should be avoided between gamma energy production cross sections defined for neutron groups 1 and 2, as given above, and gamma energy absorption cross sections, which have the same units, Mev-barn, and are defined for gamma groups 3 and 4 by Eq. 1.2 in Chapter 1). The neutron-induced gamma production data is sometimes available in the literature in this form, for each gamma-producing neutron reaction, at different neutron energies. This leaves the associated gamma spectrum unspecified. The uncertainties here, too, can be large, because of the experimental uncertainties in cross section measurements, or because of the lack of detailed data at all neutron energies. It should also be noted that the gamma production cross sections, whenever available, are not likely to have been subjected to the same degree of evaluation.

as the corresponding neutron reaction cross sections.

3. Neutron-Energy-Dependent Gamma Energy Yield, Mev, per Reaction, for each reaction, given by (using the capture reaction as an example):

$$(E_3\sigma_{13}^C + E_4\sigma_{14}^C)/\sigma_{C1}; \text{ Gamma Energy Produced per Capture in Neutron Group 1,}$$

and

$$(E_3\sigma_{23}^C + E_4\sigma_{24}^C)/\sigma_{C2}; \text{ Gamma Energy Produced per Capture in Neutron Group 2.}$$

The gamma energy produced per reaction, for other reactions, is given similarly. The uncertainties here are minimal, in most cases probably not more than 10% and even as little as 2 to 5%. The gamma production data in the literature is sometimes available in this form, i.e., per reaction, with an associated gamma spectrum, at various neutron energies. When this data is utilized, it is multiplied by the corresponding neutron reaction cross sections (σ_{C1} , σ_{C2} ; σ_{f1} , σ_{f2} , etc.) to obtain gamma production cross sections.

However, it can be seen that if the gamma production data in other forms is utilized, without further evaluation, the uncertainties that are typical of the production cross sections will propagate into the gamma energy yields per reaction. This is best illustrated with the help of a numerical example from, say, the LASL 42-group

coupled neutron-gamma cross section set. It has 30 neutron groups and 12 gamma groups. The neutron capture cross section for sodium in group 17, for example, is 0.011057 barn. Capture is the only gamma producing reaction in this group, because the group's upper energy bound is below inelastic threshold. Let ϕ_{17} be the neutron flux ($\#/\text{cm}^2\text{-sec}$) in this group, and N be the atom density ($\#/\text{cm}^3$) of sodium in the reactor. Then captures in sodium in group 17, C_{17} , are given by:

$$C_{17} = \phi_{17} \cdot N \cdot (0.011057). \quad \#/\text{cm}^3\text{-sec} \quad (3.8)$$

The gamma production cross sections for sodium, for neutron group 17, are listed in Table 3.2. The total gamma energy yield from group-17-captures in sodium is given by:

$$\sum_{j=31}^{42} \phi_{17} \cdot N \cdot \sigma_{17 \rightarrow j} \cdot E_j = 0.0081 \text{ Mev}/\text{cm}^3\text{-sec}. \quad (3.9)$$

where $\sigma_{17 \rightarrow j}$ and E_j are as in Table 3.2. Then, the gamma energy yield, E_{γ}^{17} , per group-17-neutron captured in sodium is given by:

Table 3.2

Gamma Production Cross Sections for Sodium from Neutron
Group 17 in LASL 42-Group Coupled Neutron-Gamma Cross Sec-
tion Set

Gamma Group j	Cross Section $\sigma_{17 \rightarrow j}$, barn	Gamma Group Energy, E_j , Mev
31	0.0	9.5
32	0.0	8.5
33	0.0	7.5
34	3.0918×10^{-5}	6.5
35	1.4723×10^{-4}	5.5
36	2.6501×10^{-4}	4.5
37	8.3920×10^{-4}	3.5
38	7.9503×10^{-4}	2.5
39	1.9140×10^{-4}	1.5
40	3.8279×10^{-4}	0.75
41	1.1896×10^{-3}	0.30
42	7.0375×10^{-4}	0.055

$$E_Y^{17} = \frac{\sum_{j=31}^{42} \phi_{17} \cdot N \cdot \sigma_{17 \rightarrow j} \cdot E_j}{C_{17}}$$

$$= 0.73 \text{ Mev.} \quad (3.10)$$

Thus each time a group-17-neutron is captured in sodium, only 0.73 Mev of gamma energy is introduced into the reactor. The computed gamma energy yield of 0.73 Mev deviates by a factor of ~10 from a reasonable value of 7 Mev per capture (excluding post-capture decay gammas). This is somewhat extreme, but not an exception, as will be shown in Section 3.5. It may be argued, that when an integration is performed over neutron groups, one may "hit" a reasonable energy yield value because of the "smoothing" effect of the neutron spectrum. In some cases, this is the situation, in others, it is not. This will be discussed further in Section 3.5.

From the viewpoint of compatibility in coupled neutron gamma cross section sets, inelastic-scattering gamma energy, though a small fraction of total gamma energy (see Section 3.4), is also of interest. Once the inelastic-scattering cross sections are entered as part of the neutron group cross sections, the inelastic gamma energy yields are implicitly defined by:

$$E_{\gamma, g \rightarrow g'}^{in} = \left\{ \left(\frac{A}{A+1} \right) E_g - \left(\frac{A-1}{A} \right) E_{g'} \right\}, \quad (3.11)$$

where

$E_{\gamma, g \rightarrow g'}^{in}$ is the gamma energy yield per inelastic collision from neutron group g to g' ,

E_g and $E_{g'}$ are the average neutron energies in groups g and g' , respectively, and

A is the mass number of the atom.

If $\sigma_{g \rightarrow g'}^{in}$ are the corresponding inelastic scattering cross sections, the total inelastic scattering gamma-energy production cross section, $\overline{E\sigma}_g^{in}$, Mev-barn, for neutron group g will be given by:

$$\overline{E\sigma}_g^{in} = \sum_{g'}^G \sigma_{g \rightarrow g'}^{in} \cdot E_{\gamma, g \rightarrow g'}^{in}, \quad (3.12)$$

where $E_{\gamma, g \rightarrow g'}^{in}$ are as given by Eq. 3.11, and G is the total number of neutron groups. The only choice left is with regard to the spectrum of inelastic gammas produced. (The choice of the spectrum would, in theory, affect the gamma energy yields because of the need to conserve momentum as the nuclei recoil following emission of the de-excitation gammas; however, this effect is negligible, and has not been included in Eq. 3.11. The larger recoil effects due to incident and re-emitted neu-

trons are included, however).

It will be shown in Section 3.5 that these energy balance considerations, whether for absorption or inelastic scattering events, are most likely not reflected in the two coupled neutron-gamma cross section sets used in this work as of their dates of acquisition by M.I.T. (1974). This was partly recognized by West (O-1) of Babcock and Wilcox for gamma production data for iron in the ORNL 40-Group coupled cross section set and led to some corrections (O-1). It will be shown, in Section 3.5, that a newer version still over-predicts inelastic-scattering gamma energy in the softer (reflector) neutron spectra by almost an order of magnitude. The LASL 42-group coupled neutron-gamma cross section set appears to be in a similar status (see Section 3.5).

The remainder of this chapter will proceed as follows. Section 3.2 describes the reference reactor configurations used for most of the calculations done in this chapter. In Section 3.3, the effect of gamma production spectra on gamma heating calculations is investigated. In Section 3.4, the contribution of inelastic-scattering gammas to the total gamma energy source in FBR media is estimated utilizing the group inelastic scattering cross sections from the ABBN 26-Group neutron cross section set (B-2). The effect of incident-neutron-energy on gamma energy

yields from capture and fission reactions is also estimated using approximate prescriptions. In Section 3.5, the gamma heating calculations done using ORNL and LASL coupled neutron-gamma cross section sets are compared and evaluated along the lines discussed above. In Section 3.6, the leakage of gammas from the core zone to the blanket and, from the blanket to the reflector is investigated. It is shown that the leakage of gammas contributes only a small fraction to the total gamma heating in the LMFBR blanket and reflector regions, and a simple method based upon gamma energy conservation and continuity of gamma flux at the interfaces is developed and compared with exact transport calculations. In Section 3.7, the experience obtained in this chapter is utilized to do gamma heating analysis of the Blanket Mock-up No. 4, irradiated in the Blanket Test Facility (BTF) at the M.I.T. Reactor, and the results are compared with the experimental measurements done previously by Scheinert (S-3). In Section 3.8, the importance of bremsstrahlung radiation in the reflector region of the reactor will be investigated. A summary of this chapter is included in Section 3.9.

3.2 Reference Reactor Configurations

For most of the gamma heating calculations done in this chapter, the 250 MWe LMFBR configuration shown in Fig. 3.3 was employed. Tagishi has extensively studied the breeding neutronics of this design (T-2). The main features of this reactor are: the two approximately equal-volume, oxide-fueled core zones; three oxide-fueled blanket rows, each fifteen centimeters thick, and a 50 cm thick stainless steel radial reflector. A radial one-dimensional version, shown in Fig. 3.4, of this two-dimensional reference reactor was used to facilitate the analysis performed in this chapter. Table 3.3 summarizes the physical characteristics and dimensions, and Table 3.4 gives the compositions of the various regions shown in Fig. 3.3. The isotopic number densities are listed in Table 3.5.

Figure 3.5 shows the elevation view of the upper right quadrant of a design at another extreme of interest: a large 3000 MWe oxide-fueled reactor, with six core zones, three blanket rows and a stainless steel radial reflector. In Section 3.5, the gamma heating analysis for this large reactor system will be compared with that for the smaller 250MWe reference reactor discussed above.

Fig. 3.3 Schematic Elevation View of the Upper Half of 250 MWe Reference Reactor

Fig. 3.4 One-Dimensional R-Model of the 250 MWe Reference Reactor

Table 3.3

Characteristics of the 250 MWe Reference Reactor

Parameter	Dimension	Value
Core: Height	cm	100
Diameter	cm	130
Volume	liters	1328
Axial Blanket Thickness	cm	40
Radial Blanket Thickness	cm	45
Core Averaged Power Density	KWt/liter	~450
Core Peak Power Density	KWt/liter	~670
Core Zone 1 O.D.	cm	93.6
Core Zone 1 Volume	liters	688
Core Zone 2 O.D.	cm	130
Core Zone 2 Volume	liters	640
Rated Thermal Power ^a	MWt	625
Rated Electrical Power ^b	MWe	250

^aCore plus blankets.

^bAssumed system efficiency of 40%.

Table 3.4
250 MWe Reference Reactor Parameters

	Ht. (cm)	Rad. Thick, (cm)	Percent by Volume		
			Fuel ^b	Coolant ^c	Structure ^d
Core					
Zone 1	100.0	46.8	30(85%t.d.)	50	50
Zone 2	100.0	18.2	30(85%t.d.)	50	20
Axial Blanket ^a	40.0	65.0	30(85%t.d.)	50	20
Radial blanket					
Row 1	180.0	15.0	50(95%t.d.)	30	20
Row 2	180.0	15.0	50(95%t.d.)	30	20
Row 3	180.0	15.0	50(95%t.d.)	30	20
Axial Reflector ^a					
For Axial blanket	50.0	125.0	--	50	50
For radial blanket	50.0	45.0	--	30	70
Radial Reflector	140.0	50.0	--	10	90

^aAxial blanket and reflector heights refer to thickness above or below core.

^bFuel consists of mixed uranium and plutonium dioxide in the core and uranium dioxide in the blanket. The 100 v/o, 100%t.d. molecular density is taken as 0.02447 atoms/barn-cm. Plutonium is assumed to be typical light water reactor discharge Pu at 30,000 MWD/T: 63% Pu-239/27.3%, Pu-240/5.9% Pu-241/3.8% Pu-242 (D4).

^cCoolant is sodium at ~900°F having a (100^v/o) density = 0.0220 atoms/barn-cm.

^dStructure is stainless steel with 17.7% chromium/8.3% nickel/74.0% iron having a (100^c/o) density = 0.0850 atoms/barn-cm.

Table 3.5

Isotopic Number Densities for 250MWe Reference Reactor (atoms/barn-cm)

Material	Core Zone 1	Core Zone 2	Radial Blanket	Radial Reflector
U-238	4.680×10^{-3}	4.068×10^{-3}	1.161×10^{-2}	--
Pu-239	7.565×10^{-4}	1.097×10^{-3}	0.0	--
Pu-240	3.653×10^{-4}	5.188×10^{-4}	0.0	--
Pu-241	1.441×10^{-4}	2.089×10^{-4}	0.0	--
Pu-242	5.201×10^{-4}	7.075×10^{-5}	0.0	--
U-235	1.017×10^{-5}	1.017×10^{-5}	2.326×10^{-5}	--
O-16	1.248×10^{-2}	1.248×10^{-2}	2.326×10^{-2}	--
Na	1.096×10^{-2}	1.096×10^{-2}	6.576×10^{-3}	2.192×10^{-3}
Fe	1.280×10^{-2}	1.280×10^{-2}	1.280×10^{-2}	5.737×10^{-2}
Cr	3.000×10^{-3}	3.000×10^{-3}	3.000×10^{-3}	1.366×10^{-2}
Ni	1.400×10^{-3}	1.400×10^{-3}	1.400×10^{-3}	6.375×10^{-3}

Fig. 3.5 Schematic Elevation View of the Upper Right Quadrant of a Large 3000 MWe Reactor

Table 3.6 gives the compositions of 14 representative fast reactor cores, which were used to generate neutron spectra to study spectrally-averaged gamma energy yields from ORNL and LASL coupled neutron-gamma cross section sets to be discussed in Section 3.5. The cases investigated include oxide, carbide, and metal fueled cores having compositions in the range typical of large liquid metal cooled fast breeder reactors (S-5).

3.3 The Effect of Gamma-Production Spectra on Gamma Heating Analyses

Figure 3.6 shows normalized gamma-source spectra for U-238 from neutron groups 5 and 20 in the ORNL 40-group coupled cross section set. Figure 3.7 plots the gamma source spectra for iron from neutron groups 10 and 21. It can be seen that neutron-energy-dependent gamma-source spectra have been used in the coupled cross section set. Since the uncertainty in gamma-production spectra can be large, it is of considerable interest to see the effect of spectrum shape on gamma heating calculations. For this purpose, two arbitrarily chosen gamma-production spectra, shown in Figs. 3.8 and 3.9 were introduced in the ORNL coupled neutron-gamma cross section set, without affecting the total gamma energy yield from each neutron group. The same spectra were used for all neutron groups and for all materials (i.e., oxygen, sodium,

Table 3.6

Compositions (Volume Percent) of Fast Reactor Cores
Investigated

Core Type	Volume Percent			
	PuO ₂	UO ₂	Na	Fe
Oxide Cores				
1	5	20	50	25
2	5	30	50	15
3	5	25	45	25
4	5	25	0	25
5	9	26	40	25
Carbide Cores				
1	5	25	50	20
2	5	35	35	25
3	7	18	50	25
4	5	25	0	25
5	8	22	45	25
Metal Cores				
1	5	30	50	15
2	5	25	50	20
3	5	25	0	25
4	3.5	26.5	45	25

Fig. 3.6 Normalized Gamma-Source Spectra for U-238 from Neutron Groups 5 and 20 in the ORNL 40-Group Coupled Cross Section Set

Fig. 3.7 Normalized Gamma-Source Spectra for Iron from Neutron Groups 10 and 21 in the ORNL 40-Group Coupled Cross Section Set.

Fig. 3.8 Unnormalized Gamma-Spectrum-1 Used to Study the Sensitivity of Gamma Heating Calculations to Gamma-Source Spectrum

Fig. 3.9 Unnormalized Gamma-Spectrum-2 Used to Study the Sensitivity of Gamma Heating Calculations to Gamma-Source Spectrum

iron, U-235, U-238, Pu-239 and Pu-240 needed for these calculations). This change in the gamma production spectra was effected by the use of the following equation:

$$(\sigma_{i,j}^k)^m = \frac{S_j}{\sum_{j=23}^{40} S_j E_j} \cdot \sum_{j=23}^{40} \sigma_{i,j}^k \cdot E_j, \quad (3.13)$$

where

i is the neutron group number (1 to 22),

j is the gamma group number,

k is the material,

$\sigma_{i,j}^k$ is the gamma production cross section from neutron group i to gamma group j , for material k , as in the ORNL coupled cross section set,

$(\sigma_{i,j}^k)^m$ is the corresponding modified gamma production cross section incorporating the new gamma-production spectrum,

E_j is the average gamma energy in group j , and

S_j are the ordinates as shown in Fig. 3.8 (or Fig. 3.9).

It can be seen that the total gamma energy yield from each neutron group for each material remains the same as

in the original coupled cross section set. Only the gamma production spectra are changed.

Table 3.7 compares the gamma heating rates calculated for the reference reactor, using:

1. Gamma production spectra as in the ORNL 40-group coupled cross section set,
2. A gamma production spectrum as in Fig. 3.8, and
3. A gamma production spectrum as in Fig. 3.9.

It can be seen that gamma heating calculations in FBR media are insensitive to a large extent to gamma production spectra. The maximum difference in gamma heating rate is found to be at the core-blanket interface and is an insignificant $\pm 4\%$. Note, furthermore, that the spectra of Figs. 3.8 and 3.9 provide a very strict test of the sensitivity of gamma heating calculations to the gamma production spectra. The spectrum of Fig. 3.8 introduces gammas mainly in the middle of the gamma energy range, and the spectrum of Fig. 3.9 introduces almost all gammas at very low or very high energies.

3.4 Contribution of Neutron-Energy to Gamma Energy Source

3.4.1 Inelastic-Scattering Gamma Energy

The inelastic-scattering gamma-energy source at each spatial point in the 250 MWe reference reactor of Fig. 3.4 was obtained using the 26-group ABBN cross section

Table 3.7

The Effect of Gamma-Production Spectra on Gamma Heating Calculations
in the 250MWe Reference Reactor

Region/Dis- tance from core center [†] , cm	Gamma Production Spectra as in					
	ORNL 40-Group Coupled Library	Figure 3.8	Figure 3.9	ORNL 40-Group Coupled Library	Figure 3.8	Figure 3.9
	Gamma Heating Rate, KWT/liter*			Ratio of Gamma Heating to Fission Heating		
Core Zone 1: 0.0	94.1	94.3	94.2	0.140	0.140	0.140
23.0	89.1	89.3	89.2	0.140	0.140	0.140
45.0	76.3	77.1	77.2	0.142	0.144	0.144
Core Zone 2: 49.0	82.5	81.6	81.2	0.121	0.119	0.119
61.0	64.5	63.8	63.4	0.121	0.120	0.119
63.0	53.2	51.9	51.5	0.118	0.115	0.114
Blanket Row 1: 67.0	26.2	27.6	28.4	0.679	0.715	0.737
71.0	17.2	17.7	18.0	0.814	0.837	0.851
79.0	8.90	9.00	9.05	1.250	1.263	1.270
Blanket Row 2: 85.0	6.47	6.53	6.56	1.500	1.512	1.519
91.0	4.69	4.72	4.74	1.750	1.762	1.770
93.0	3.33	3.40	3.42	1.988	2.001	2.010

(Continued)

Table 3.7(Continued)
The Effect of Gamma-Production Spectra on Gamma Heating Calculations in the 250MWe Reference Reactor

Region/Distance from core-center, [†] cm	Gamma Production Spectra as in					
	ORNL 40-Group Coupled Library	Fig. 3.8	Fig. 3.9	ORNL 40-Group Coupled Library	Fig. 3.8	Fig. 3.9
	Gamma Heating Rate, kwt/liter*			Ratio of Gamma Heating to Fission Heating		
Blanket Row 3: 99.0	2.06	2.07	2.08	2.307	2.322	2.332
105.0	1.29	1.29	1.30	2.617	2.626	2.638
109.0	1.05	1.03	1.03	2.958	2.923	2.929
Reflector: 111.0	0.314	0.317	0.320	--	--	--
115.0	0.242	0.251	0.254	--	--	--
125.0	0.184	0.186	0.186	--	--	--
135.0	0.069	0.069	0.069	--	--	--

* Normalized to core peak power density of 670 kwt/liter (~450 kwt/liter core averaged power density).

[†] Mesh intervals of 2 cm were used for these calculations. The mesh-points given in this table were chosen to show the accentuated effect near interfaces.

set (B-2) as follows:

$$E_{\gamma}^{in}(r) = \sum_k \sum_{g=1}^{26} \overline{E\sigma}_g^{in,k} \cdot N^k \cdot \phi_g(r), \quad (3.14)$$

where

$E_{\gamma}^{in}(r)$ is the inelastic gamma energy source at point r , Mev/cm³-sec,

k denotes the material,

g denotes the neutron group,

$\overline{E\sigma}_g^{in,k}$ is as defined by Eq. (3.12), for each group g , and for each material k ,

N^k is the number density of material k , as in

Table 3.5, and

$\phi_g(r)$ is the neutron group flux at point r .

The fluxes, $\phi_g(r)$, were determined by S_8 26-group transport calculations using ANISN. To obtain the inelastic-scattering gamma energy as a fraction of the total gamma energy source in the reactor, the gamma-energies from fission and capture reactions were calculated as follows:

$$E_{\gamma}^f(r) = \sum_k \sum_{g=1}^{26} E_f^k \cdot \sigma_{f,g}^k \cdot N^k \cdot \phi_g(r), \quad (3.15)$$

$$E_Y^C(r) = \sum_k \sum_{g=1}^{26} E_C^k \cdot \sigma_{C,g}^k \cdot N^k \cdot \phi_g(r), \quad (3.16)$$

$$E_Y^T(r) = E_Y^f(r) + E_Y^C(r) + E_Y^{in}(r), \quad (3.17)$$

$$F_Y^{in}(r) = E_Y^{in}(r)/E_Y^T(r), \quad (3.18)$$

where

$E_Y^f(r)$ is the gamma energy from fissions, at point r ,
Mev/cm³-sec,

$E_Y^C(r)$ is the gamma energy from captures, at point
 r , Mev/cm³-sec,

$E_Y^T(r)$ is the total gamma energy source, at point r ,
Mev/cm³-sec,

$\sigma_{f,g}^k$ is the microscopic fission cross section for
material k , group g , barns,

σ_C^k is the microscopic capture cross section for
material k , group g , barns,

$F_Y^{in}(r)$ is the inelastic gamma energy source as a
fraction of the total gamma energy source, and

E_f^k and E_C^k are the gamma energy yields, Mev, per
fission and per capture, respectively,
as listed in Table 3.8 (K-1).

Figure 3.10 plots $F_Y^{in}(r)$ as a function of the radius,
 r . It is seen that the contribution of inelastic gammas
is approximately 10 to 12 percent of the total gamma energy
source in the core-region. In the blanket-row-1, near

Table 3.8

Gamma Energy Yields Per Gamma Production Event (Mev)

Isotope	Gamma Energy from Nonfission Neutron Capture	Gamma Energy from Fission + Fission Product Decay
O-16	4.14	--
Na-23	6.96	--
Fe	7.77	--
U-235	6.54	15.2
U-238	4.80	15.9
Pu-239	6.53	13.8
Pu-240	5.24	15.0

Fig. 3.10 Contribution of Inelastic-Scattering Gammas to Total Gamma-Energy Source in the 250 MWe Reference Reactor

the core-blanket interface, it can be as much as 10 to 12 percent, but decreases to 3 to 4 percent towards the outer boundary of row-1. In blanket-row-2, the average contribution of inelastic gammas is approximately 2% and it is less than 1% in the third row of the blanket. Near the blanket-reflector interface, the inelastic-scattering gamma energy is approximately 5%, and the relative contribution again falls off as one moves deeper into the region.

Figure 3.11 compares the inelastic gamma-energy source with the neutron-heating rate in the reactor. The neutron heating rate is calculated using the following equations (B-1):

$$N_H^e(r) = \sum_{k,g} E_g \cdot (1 - e^{-\xi_g}) \cdot \sigma_{e,g}^k \cdot N^k \cdot \phi_g(r), \quad (3.19)$$

$$N_H^{in}(r) = \sum_{k,g,g'} \left\{ E_g \left(\frac{1}{A^{k+1}} \right) + E_{g'} \left(\frac{1}{A^k} \right) \right\} \cdot \sigma_{g \rightarrow g'}^{in,k} \cdot N^k \cdot \phi_g(r), \quad (3.20)$$

$$N_H^c(r) = \sum_{k,g} \left(\frac{1}{A^{k+1}} \right) \cdot \sigma_{c,g}^k \cdot N^k \cdot \phi_g(r), \quad (3.21)$$

and

Fig. 3.11 Comparison of Inelastic-Scattering Gamma-Energy and Neutron-Heating in the 250 MWe Reference Reactor

$$N_H^T(r) = N_H^e(r) + N_H^{in}(r) + N_H^c(r), \quad (3.22)$$

where

N_H denotes neutron heating rate, Mev/cm³-sec,

ξ_g is the average increase in lethargy per elastic collision in group g,

A^k denotes the mass number of the material k,

e denotes elastic scattering,

$N_H^T(r)$ is the total neutron heating rate, Mev/cm³-sec, and other symbols are the same as in the preceding equations.

It is seen that the inelastic-scattering gamma heating rate is a factor of about 1.7 more than the neutron heating rate in the core region. In the blanket, inelastic-gamma heating is comparable to neutron heating, and in the reflector, neutron heating is somewhat more than inelastic gamma heating.

3.4.2 Effect of Incident-Neutron Energy on Gamma Energy Yields from Capture and Fission Reactions.

The effect of the incident neutron-energy on gamma energy yields from capture and fission reactions can be estimated using the following equations:

$$E_Y^C(r) = \sum_{g,k} \left(E_g \cdot \frac{A^k}{A^{k+1}} + E_{c,k}^{th} \right) \cdot \sigma_{c,g}^k \cdot N^k \cdot \phi_g(r), \quad (3.23)$$

and

$$E_Y^f(r) = \sum_{g,k} \left(\frac{E_g + T}{T} \right) \cdot E_{f,k}^{th} \cdot \sigma_{f,g}^k \cdot N^k \cdot \phi_g(r), \quad (3.24)$$

where

$E_Y^C(r)$ is the gamma energy source, Mev/cm³-sec at point r in the reactor,

$E_Y^f(r)$ is the gamma energy source, Mev/cm³-sec from fission reactions at point r in the reactor,

E_g is the midpoint energy, Mev, of neutron group g ,

T is the total energy released in fission (~ 207 Mev),

$E_{c,k}^{th}$ is the gamma energy, Mev, released per thermal capture in material k (Table 3.8),

$E_{c,k}^f$ is the gamma energy, Mev, released per thermal fission in material k (Table 3.8),

and other symbols are as in the preceding equations of this section.

Equation 3.23 is exact. In Eq. 3.24, the additional gamma energy due to non-thermal fissions is approximately accounted for by the factor $(E_g + T)/T \cdot (K-1)$. Figure 3.12 shows the contribution of the incident neutron energy to the gamma energy source from capture and fission reactions (i.e. the effect of including the term $E_g \cdot A^k/(A^{k+1})$ in Eq. 3.23 and the factor $(E_g + T)/T$ in Eq. 3.24). It can be

Fig. 3.12 Contribution of Incident-Neutron-Energy to Gamma-Energy Source from Capture and Fission Reactors in 250 MWe Reference Reactor

seen that if energy release is computed as if all captures and fissions were caused by thermal neutrons, the error introduced in the calculation of the gamma energy source is not more than ~1%. This should be compared with the approximately 5 to 10% error introduced in the calculation of the gamma energy source distribution if the inelastic-scattering gamma energy is neglected (as is more often done; see for example Ref. K-1).

Figure 3.13 shows the total contribution of neutron-energy (via inelastic scattering and additional gamma energy in captures and fissions) to the total gamma energy source in the 250 MWe reference reactor, and Table 3.9 summarizes the gamma energy sources from capture, fission and inelastic-scattering reactions.

The overall conclusion of these calculations is that the total contribution of neutron-energy to the gamma energy source in fast reactor environments is of the order of 10% in harder (core) neutron spectra (mainly via inelastic scattering), and lesser in softer (blanket and reflector) spectra.

Thus any sophisticated model for gamma heating calculations, (for example, coupled neutron-gamma cross section sets) must be justified in part on the basis of its capacity to calculate accurately this additional (~10%) gamma energy source in the reactor. (It may also be

Fig. 3.13 Total Contribution of Neutron-Energy (via Inelastic Scattering and Additional Gamma-Energy in Captures and Fissions) to Gamma-Energy Source Distribution in 250 MWe Reference Reactor

Table 3.9

Summary of Gamma-Energy Sources in 250 MWe Reference Reactor Calculated
Using 26-Group ABBN Cross Section Set

Region	Percent Contribution to Total Gamma-Energy Source							
	Mat- erial Reac- tion	O	Na	Fe, Cr, Ni	U-235 U-238	Pu-239 Pu-241	Pu-240 Pu-242	Total
Core Zones 1 & 2	Fission	-	-	-	6.5	51.0	4.6	62.0
	Capture	0.3	0.3	2.1	14.7	6.4	3.6	27.4
	In. Scat.	0.04	1.4	4.7	3.9	0.6	0.3	10.7
	Total	0.3	1.7	6.8	25.1	58.0	8.4	100.0
Blanket Row 1	Fission	--	--	--	8.9	--	--	8.9
	Capture	0.2	0.3	2.7	80.7	--	--	83.9
	In. Scat.	0.03	0.4	2.1	4.8	--	--	7.3
	Total	0.3	0.7	4.8	94.3	--	--	100.0
Blanket Row 2	Fission	--	--	--	3.7	--	--	3.7
	Capture	0.1	0.2	2.3	91.5	--	--	94.1
	In. Scat.	0.01	0.1	0.6	1.4	--	--	2.2
	Total	0.1	0.4	2.9	96.6	--	--	100.0

(continued)

Table 3.9 (Continued)
Summary of Gamma-Energy Sources in 250 MWe Reference Reactor Calculated
Using 26-Group ABBN Cross Section Set

Region	Percent Contribution to Total Gamma-Energy Source							
	Mat- erial Reac- tion	O	Na	Fe, Cr, Ni	U-235 U-233	Pu-239 Pu-241	Pu-240 Pu-242	Total
Blanket Row 3	Fission	--	--	--	2.4	--	--	2.4
	Capture	0.03	0.2	2.0	94.4	--	--	96.7
	In Scat.	0.01	0.1	0.3	0.6	--	--	1.0
	Total	0.04	0.3	2.3	97.5	--	--	100.0
Distance into Reflector,								
5cm	Capture	--	0.8	95.7	--	--	--	96.5
	In. Scat.	--	0.1	3.4	--	--	--	3.5
	Total	--	0.9	99.1	--	--	--	100.0
10cm	Capture	--	0.8	97.3	--	--	--	98.1
	In. Scet.	--	--	1.9	--	--	--	1.9
	Total	--	0.8	99.2	--	--	--	100.0

noted that the calculation of the inelastic-gamma energy source does not inherently require "coupled" neutron-gamma cross sections). It will be brought out in the next section that the two coupled neutron-gamma cross section sets used in this study, are presently inadequate for this aspect of gamma heating calculations, and in fact, introduce much larger uncertainties in the calculation of the rest (90%) of the gamma energy source.

3.5 Gamma Heating Analyses Using ORNL and LASL Coupled Neutron-Gamma Cross Section Sets

3.5.1 Gamma Heating Analysis for the 250 MWe Reference Reactor

Figure 3.14 shows the gamma heating rate in the core, blanket and reflector regions of the 250 MWe reference reactor calculated using the ORNL 40-group coupled neutron gamma cross section set. Figure 3.15 compares the gamma heating rate with the fission heating rate in the core and the blanket regions, and Fig. 3.16 compares the gamma heating rate with the neutron heating rate in the reflector region of the reactor.

It can be seen that in blanket-row-3, gamma heating is ~2.5 times fission heating). In the reflector, external to the blanket, gamma heating constitutes more than 95% of the total heating. (Note that here and elsewhere in this report the blanket calculations are for the beginning-of-life conditions, prior to build-up

Fig. 3.14 Gamma Heating Rate in 250 MWe Reference Reactor Calculated Using ORNL Coupled Cross Section Set

Fig. 3.15 Ratio of Gamma Heating Rate to Local Fission Power in 250 MWe Reference Reactor

Fig. 3.16 Ratio of Gamma Heating to Neutron Heating
in Steel Reflector of 250 MWe Reactor

of bred plutonium).

Table 3.10 summarizes the gamma energy sources and sinks in the various regions of the reference reactor. It can be seen that in the core and the blanket regions, the coolant and the structure contribute less than 10% to the gamma energy source, but absorb up to 30% of the gamma energy. In the reflector region, coolant, here assumed to be 10 volume percent of the region, contributes only ~1% to the gamma energy source and sink.

3.5.2 Gamma Heating Analysis for a 3000 MWe Reactor System

Figure 3.17 compares the gamma heating rates for the 3000 MWe reactor calculated using the ORNL and the LASL coupled neutron-gamma cross section sets, normalized to the same core peak power density of 700 KWt/liter. Figure 3.18 compares the ratios of the gamma heating rate to the fission heating rate. It can be seen that in the core and the reflector regions, the difference in gamma heating rates obtained from the two coupled cross section sets is approximately 60%. In the blanket-row-1, the difference in gamma heating rates is ~100%; in the blanket-rows-2 and 3, the difference is less (~15%).

We may also note that on the whole there appear to be no profound differences between the behavior of gammas in the 3000 MWe and 250 MWe designs (e.g. compare Fig. 3.15

Table 3.10

Summary of Gamma-Energy Sources and Sinks in 250 MWe Reference Reactor
Calculated Using 40-Group ORNL Coupled Cross Section
Set

Region		Percent Contribution to Total Gamma-Energy Source/Sink						Total
		0	Na	Fe, Cr, Ni	U-235 U-238	Pu-239 Pu-241	Pu-240 Pu-242	
Core Zones 1&2	Source	0.03	2.0	7.0	25.0	52.0	14.0	100.0
	Sink	4.0	5.0	20.0	48.0	16.0	7.0	100.0
Blanket Row 1	Source	0.03	1.2	7.5	91.3	--	--	100.0
	Sink	5.0	2.0	13.0	80.0	--	--	100.0
Blanket Row 2	Source	--	1.0	7.0	92.0	--	--	100.0
	Sink	5.0	2.0	13.0	80.0	--	--	100.0
Blanket Row 3	Source	--	1.0	6.0	93.0	--	--	100.0
	Sink	5.0	2.0	14.0	79.0	--	--	100.0
Steel Reflector	Source	--	0.7	99.3	--	--	--	100.0
	Sink	--	1.0	99.0	--	--	--	100.0

Fig. 3.17 Gamma Heating Rate in a 3000 MWe Reactor

Fig. 3.18 Ratio of Gamma Heating Rate to Local Fission Power in 3000 MWe Reactor

and 3.18). Hence we may conclude that reactor size is not an important consideration in the present evaluation.

3.5.3 Evaluation of the Gamma Heating Analysis

In the preceding section, we compared the gamma heating rates for the 3000 MWe reactor system, as calculated by the ORNL and the LASL coupled neutron-gamma cross section sets. However, a better comparison and evaluation of the gamma energy yield data in the two coupled cross section sets is provided by looking at each material individually. As was shown in Section 3.4, approximately 90% (or more) of the gamma energy comes from fission and capture reactions in fast reactor media. This is also roughly true for fissile or fertile materials individually, although inelastic scattering gamma energy from U-238 in harder core spectra can be somewhat more than 10% of the capture and fission gamma-energy from U-238 (see Table 3.9). In the blanket spectra, however, where U-238 contributes more than 90% to the gamma energy source (at the beginning of life), the inelastic-scattering gamma energy (from U-238 or total) is of the order of 5%.

In view of the above considerations an approximate check on the gamma energy yield data in the coupled neutron-gamma cross section sets can be achieved by considering only capture and fission contributions, in the form:

$$E_T^k \approx E_C^k \cdot C^k + E_f^k \cdot F^k, \quad (3.25)$$

where

$$E_T^k = \sum_{i=1}^N \sum_{j=1}^G \phi_i \cdot \sigma_{i \rightarrow j}^k \cdot N^k \cdot E_j, \quad (3.26)$$

$$F^k = \sum_{i=1}^N \phi_i \cdot \sigma_{f,i}^k \cdot N^k, \quad (3.27)$$

$$C^k = \sum_{i=1}^N \phi_i \cdot \sigma_{c,i}^k \cdot N^k, \quad (3.28)$$

E_C^k is a constant, Mev, approximately equal to the gamma-energy yield per capture in material k,

E_f^k is another constant, Mev, approximately equal to the gamma-energy yield per fission in material k,

i denotes the neutron group,

j denotes the gamma group

ϕ denotes the neutron flux, and

$\sigma_{i \rightarrow j}^k$ are the gamma production cross sections.

Thus a plot of E_T^k/C^k versus F^k/C^k should be approx-

imately linear for U-235, U-238, Pu-239 and Pu-240. To test this hypothesis a wide variety of cases, involving both hard and soft neutron spectra, were generated by calculating the 250 MWe and the 3000 MWe reactors shown in Figs. 3.4 and 3.5, using in succession the oxide, carbide and metal core compositions listed in Table 3.6.

Plots of the resulting ratios are shown in Figs. 3.19 to 3.22. In these figures, the points at higher (spectral) fission to capture ratios, F^k/C^k , correspond to the metal core compositions, and the points at lower values of F^k/C^k correspond to the softer blanket spectra. The constants E_C^k and E_f^k as obtained from these plots are given in Table 3.11. The difference in the gamma energy yield data in the two coupled cross section sets is clearly brought out in these figures. In fact, the values of the constants E_C^k and E_f^k obtained from these plots (see Table 3.11) deviate from expected gamma energy yield values per capture and per fission (see Table 3.8) to an extent that the linearity of these plots may in fact be mainly due to a fortuitous smoothing effect introduced by averaging over the neutron spectra.

Thus the gamma energy yield data in the ORNL and the LASL coupled neutron-gamma cross section sets does not appear to conform to physically realistic limits. This is further brought out in Tables 3.12 to 3.14. For

Fig. 3.19 Plot of Equation 3.25 for U-235

Fig. 3.20 Plot of Equation 3.25 for U-238

Fig. 3.21 Plot of Equation 3.25 for Pu-239

Fig. 3.22 Plot of Equation 3.25 for Pu-240

Table 3.11

The Values of E_c^k and E_f^k Obtained from Figures 3.19 to 3.22

Material	E_c^k , Mev		E_f^k , Mev	
	ORNL Coupled Cross Section Set	LASL Coupled Cross Section Set	ORNL Coupled Cross Section Set	LASL Coupled Cross Section Set
U-235	1.6 [†]	8.7	15.9	6.7
U-238	6.3	7.0	32.1	17.3
Pu-239	2.9	7.0	16.2	7.1
Pu-240	13.2	11.0	33.9	5.0

[†]These values should be compared with gamma energy yields per capture and per fission given in Table 3.8.

Table 3.12

Neutron-Induced Gamma Energy Yields from Pu-240 in LASL
Coupled Neutron Gamma Cross Section Set (Neutron Groups
22 to 30*)

Neutron Group g	Neutron Energy Range in Group g	Gamma Energy Yield per capture in Group g, Mev
22	0.454-0.167 kev	5.611
23	167-61.4 ev	5.631
24	61.4-22.6 ev	2.617
25	22.6-8.32 ev	5.727
26	8.32-3.06 ev	1.745
27	3.06-1.13 ev	24.29
28	1.13-0.414 ev	758.8
29	0.414-0.152 ev	9.297
30	Thermal	5.544

* Neutron energies in these groups are below inelastic and fission thresholds; thus capture is the only gamma-producing reaction. Any significant variation from one group to the other is essentially the uncertainty in gamma production cross section data or an error. A reasonable expected value of gamma energy yield per capture in all these groups would be ~5.24 Mev (see Table 3.8).

Table 3.13

Neutron-Induced Gamma Energy Yields from U-238 in ORNL
Coupled Neutron-Gamma Cross Section Set (Neutron Groups
15 to 22^a)

Neutron Group g	Neutron Energy Range in Group g	Gamma Energy Yield Per Capture in Group g, Mev	
		Old Version ^b	New Version
15	3.35-0.583 kev	7.062 ^c	Same as in Old Version
16	0.583-0.101 kev	4.959	
17	101-29 ev	7.320	
18	29-10.7 ev	9.685	
19	10.7-3.06 ev	14.223	
20	3.06-1.12 ev	4.473	
21	1.12-0.414 ev	4.480	
22	Thermal	4.489	

^aNeutron energies in these groups are below inelastic and fission thresholds; thus capture is the only gamma-producing reaction. Any significant variation of gamma energy yield per capture in these groups is essentially the uncertainty in gamma production cross section data, ~ a factor of 3 here, which affects gamma energy yield by the same factor; actual uncertainty in gamma energy yield is probably 5 to 10%.

^bThis version was used for most of the work in this report.

^cA reasonable value of gamma energy yield per capture in all these groups would be ~4.8 Mev (see Table 3.8).

Table 3.14

Neutron-Induced Gamma Energy Yields from Pu-239 in ORNL Coupled Neutron Gamma Cross Section Set (Neutron Groups 15 to 22^a)

Neutron Group g	Gamma Energy Yields per Absorption in Group g, Mev	
	Old Version ^b	New Version
15	10.258	10.258
16	11.309	11.309
17	9.597	9.597
18	10.768	10.539
19	20.768 ^c	20.768
20	13.207	13.207
21	2.512 ^c	2.512
22	1.504x10 ⁻⁵	12.176

^a Neutron energies in these groups are below inelastic threshold; thus absorption (capture and fission) is the only gamma producing reaction.

^b This version was used for most of the work in this report.

^c Fission to capture ratio in both these groups is ~2; yet the gamma energy yield per absorption, which should be approximately the same, differs by a factor of 8.

example, one absorption in neutron group 26 (LASL coupled cross section set) in Pu-240 gives only 1.745 Mev of gamma energy; an absorption in group 27 gives 24.29 Mev of gamma energy; an absorption in group 28 gives 758.8 Mev of gamma energy. Such inconsistencies are frequent enough to make the gamma production cross section data unreliable.

The gamma production cross section data for iron and sodium exhibit similar inconsistencies. This is brought out in Table 3.15 for the LASL coupled cross section set and in Table 3.16 for the ORNL coupled cross section set. The gamma energy yields from iron are of special interest, because in the stainless steel reflector of the fast reactor, almost all gamma heating is due to gamma energy from captures in iron (plus those in Cr and Ni). In Tables 3.15 and 3.16, the gamma energy yields from neutron groups whose upper energy bounds are below inelastic threshold, are given. The gamma energy yields from other neutron groups cannot be examined in a similar fashion because of the possibility that from a particular neutron group above inelastic threshold, the inelastic-scattering gamma energy can be the dominant fraction of the total gamma energy yield. However, the overall inconsistency is brought out in Fig. 2.23, where the ratio of the inelastic scattering gamma energy to

Table 3.15

Neutron-Induced Gamma Energy Yields from Fe and Na in LASL Coupled
Neutron-Gamma Cross Section Set (Neutron Groups 16 to
23^a)

Neutron Group g	Gamma Energy Yield per Capture in Group g, Mev	
	Fe	Na
16	7.65 ^b	1.03 ^c
17	7.90	0.73
18	7.94	13.30
19	76.13	3.55
20	360.30	4.04
21	8.21	3.99
22	6.86	4.33
23	7.58	5.22

^a Neutron energies in these groups are below inelastic threshold; thus capture is the only gamma-producing reaction. Any significant variation of gamma energy yield per capture in these groups is essentially the uncertainty in gamma production cross section data or an error.

^b A reasonable value of gamma energy yield per capture in all these groups for Fe would be ~7.8 Mev (see Table 3.8).

^c A reasonable value of gamma energy yield per capture in all these groups would be ~7 Mev, excluding post-capture decay gammas (see Table 3.8).

Table 3.16

Neutron-Induced Gamma Energy Yields from Fe in ORNL Coupled Neutron-Gamma Cross Section Set (Neutron Groups 14 to 22*)

Neutron Group g	Gamma Energy Yield per Capture in Group g, Mev	
	Old Version	New Version
14	10.161	7.723
15	27.068	7.685
16	7.782	7.548
17	7.840	7.549
18	7.525	7.539
19	7.466	7.537
20	7.556	7.551
21	7.559	7.550
22	31.734	7.513

* Neutron energies in these groups are below inelastic threshold;
thus capture is the only gamma-producing neutron reaction.

Fig. 3.23 Inelastic Gamma-Energy as a Fraction of Capture Gamma-Energy in the Steel Reflector of the 250 MWe Reference Reactor

the capture gamma energy from iron is plotted for the reflector region of the 250MWe reference reactor, as calculated by:

1. The LASL coupled cross section set (obtained in December 1974),
2. The older version of the ORNL coupled cross section set (obtained in 1972),
3. The newer version of the ORNL coupled cross section set (obtained in June 1975), and
4. The 26-group ABBN cross section set (B-2) based upon energy balance considerations (see Section 3.4).

The inelastic-scattering gamma energy in the coupled neutron-gamma cross section sets was estimated by computing the excess energy over that attributable to capture, as follows:

$$\frac{E_Y^{in}(r)}{E_Y^c(r)} = \frac{\sum_{i,j} E_j \cdot \sigma_{i \rightarrow j}^{Fe} \cdot N^{Fe} \cdot \phi_i(r)}{\sum_i E_c^{Fe} \cdot \sigma_{c,i}^{Fe} \cdot N^{Fe} \cdot \phi_i(r)} - 1, \quad (3.29)$$

where

i denotes the neutron group,

j denotes the gamma group,

c denotes capture,

in denotes inelastic scattering,

E_j is the mean energy of gamma group j ,

E_C^{Fe} is the gamma energy yield per capture in iron,
~7.8 Mev,

and other symbols have their usual meaning.

It can be seen in Fig. 3.23 that the coupled neutron-gamma cross section sets highly over-predict the inelastic scattering gamma energy in the reflector region. It may be noted that the corrections made in the newer version of the ORNL coupled cross section set in the iron data (see Table 3.16) have not fully resolved this inconsistency; the inelastic-scattering gamma energy is still over-predicted by an order of magnitude. This indicates lack of energy-balance considerations in groups above inelastic threshold. It may be pointed out here that in the above estimation, any discrepancies in gamma energy yields below inelastic threshold will also be lumped with E_Y^{in} . However, in the new version of the ORNL 40-group coupled cross section set, there are almost no discrepancies in gamma energy yields from neutron groups below inelastic threshold in iron data (see Table 3.16); still it over-predicts the E_Y^{in} by almost an order of magnitude as indicated above.

The overall conclusion of this section, therefore, is that at the present state of coupled neutron-gamma cross section set development, the neutron-induced gamma-energy source is more reliably estimated by calculating the number of neutron captures and fissions in each material

at each spatial point as accurately as possible, and then multiplying them by the gamma energy yields per capture and per fission for each material. Approximately 90% of the gamma energy source in the core region, and more than 95% of the gamma energy source in the blanket and the reflector regions is accurately calculated by this method. The inelastic-scattering gamma energy is also more reliably estimated based on neutron energy balance considerations. Gamma energy from other reactions (post-capture decay gammas; (n,p) , $(n, 2n)$ reactions, etc.) can also be conveniently included in a similar manner.

3.6 Leakage of Gamma Energy Across Interfaces

Figure 3.24 shows the ratio of gamma heating (gamma energy deposition) to the local gamma energy source at each spatial point in the 250 MWe reference reactor, as calculated using the ORNL 40-group coupled cross section set. It can be seen that leakage from core-zone-2 to core-zone-1 increases the gamma heating rate near the interface by $\sim 10\%$. The leakage gamma energy is deposited within the first few transport-mean-free-paths of the boundary and accounts for approximately 3 to 4% additional gamma heating in this region. Leakage from core-zone-2 to blanket-row-1 can increase the gamma heating rate near the interface by a factor of ~ 1.6 and accounts for $\sim 10\%$ additional gamma heating in the first 6 to 8 cm of the blanket-row-1. Leakage from blanket-row-3 to the reflector increases the gamma heating rate in the reflector by a factor ~ 2 near the interface and accounts for $\sim 20\%$ additional gamma heating in the first 8 to 10 centimeters of the reflector region.

In this section, a simple method to estimate the leakage of gammas, based upon a gamma energy balance and the continuity of gamma flux at the interface, will be developed and compared with "exact" 18-gamma-group (using the ORNL coupled cross section set) S_8P_1 transport calculations. Towards this goal Fig. 3.25 depicts, schematically the gamma energy leakage in the vicinity of

Fig. 3.24 Local Gamma-Energy Deposition-to-Source Ratio in the 250 MWe Reference Reactor

Fig. 3.25 Schematic Depiction of Gamma Energy Leakage at an Interface

an interface. The following approximate relations are assumed to apply near the interface:

$$E_{d1}(r_1) = E_{s1}(r_1) - \Delta E_1 \left(\frac{r_i}{r_1} \right) \exp(-\Delta r_1 / \lambda_1), \quad (3.30)$$

and

$$E_{d2}(r_2) = E_{s2}(r_2) + \Delta E_2 \left(\frac{r_i}{r_2} \right) \exp(-\Delta r_2 / \lambda_2), \quad (3.31)$$

where

$$\Delta r_1 = r_i - r_1, \quad (3.32)$$

$$\Delta r_2 = r_2 - r_i, \quad (3.33)$$

λ denotes the one-group gamma transport-mean-free-path (Table 2.2),

E_s denotes gamma energy source,

E_d denotes gamma energy deposition, and other

symbols are as depicted in Fig. 3.25. The two unknowns, ΔE_1 and ΔE_2 , in Eqs. 3.30 and 3.31 are determined by using a gamma energy balance and continuity of gamma flux at the interface:

$$\int_{r_i}^{\infty} \{E_{s1}(r_1) - E_{d1}(r_1)\} \cdot 2\pi r_1 \cdot dr_1 = \int_{r_i}^{\infty} \{E_{d2}(r_2) - E_{s2}(r_2)\} \cdot 2\pi r_2 \cdot dr_2, \quad (3.34)$$

and

$$\frac{E_{d1}(0)}{\overline{\Sigma}_{a1}} = \frac{E_{d2}(0)}{\overline{\Sigma}_{a2}} = \phi_i^Y, \quad (3.35)$$

where

ϕ_i^Y is the gamma flux at the interface, and $\overline{\Sigma}_{a1}$ and $\overline{\Sigma}_{a2}$ are the macroscopic gamma energy absorption cross sections obtained from the one-group microscopic cross sections given in Table 3.17. Use of Eqs. 3.34 and 3.35 in Eqs. 3.30 and 3.31 leads to:

$$\Delta E_1 = \frac{\overline{\Sigma}_{a2} \cdot E_{s1}(0) - \overline{\Sigma}_{a1} \cdot E_{s2}(0)}{\overline{\Sigma}_{a1} \cdot (\lambda_1/\lambda_2) + \overline{\Sigma}_{a2}}, \quad (3.36)$$

and

$$\Delta E_2 = \frac{\overline{\Sigma}_{a2} \cdot E_{s1}(0) - \overline{\Sigma}_{a1} \cdot E_{s2}(0)}{\overline{\Sigma}_{a1} + \overline{\Sigma}_{a2} (\lambda_2/\lambda_1)}. \quad (3.37)$$

In Tables 3.18 to 3.20, the gamma heating rates calculated using the above equations are compared with those obtained using the ORNL coupled cross section set. It is seen that the method outlined above can predict gamma heating rates at the interfaces, where material and source discontinuities are encountered, with reasonable accuracy. The maximum error in gamma heating rates is found to be

Table 3.17

One Group Microscopic Gamma Energy-Absorption Cross Sections*

<div> <div>Mat- erial</div> <div>Region</div> </div>	One Group Microscopic Gamma Energy Absorption Cross Section, $\bar{\sigma}_a$, Mev-barn				
	O	Na	Fe	U	Pu
Core	0.65	0.90	2.25	23.5	24.5
Blanket	0.63	0.86	2.18	18.5	--
Reflector	--	0.73	2.15	--	--

*Obtained by collapsing 18-group microscopic gamma-energy absorption cross sections over representative gamma spectra appropriate to the core, blanket and reflector regions of the 250 MWe reference reactor.

Table 3.18

Gamma-Energy Deposition near the Core-Zone-2/Core-Zone-1
Interface in the 250 MWe Reference Reactor

Region/Dis- tance from Interface, cm	Gamma Heating Rate, KWt/liter*		Percent Error
	Exact 18-Group Gamma Transport Calculations	Approximate Method Eqs. 3.30 3.31	
Core Zone 1			
5.5	75.25	75.36	+ 0.1
4.5	74.53	74.73	+ 0.3
3.5	73.86	74.26	+ 0.5
2.5	73.44	74.12	+ 0.9
1.5	73.23	74.54	+ 1.8
0.5	74.47	75.58	+ 1.5
Core Zone 2			
0.5	77.23	77.48	+ 0.3
1.5	78.22	78.35	+ 0.2
2.5	77.62	78.25	+ 0.8
3.5	76.67	77.54	+ 1.1
4.5	75.37	76.43	+ 1.4
5.5	73.90	75.04	+ 1.5

* Normalized to core-peak-power density of 670 KWt/liter.

Table 3.19

Gamma-Energy Deposition near the Core-Zone-2/Blanket-Row-1
Interface in the 250 MWe Reference
Reactor

Region/Dis- tance from Interface, cm	Gamma Heating Rate, KWt/liter*		Per- cent Error
	Exact 18-Group Gamma Transport Calculations	Approximate Method Eqs. 3.304 & 3.31	
Core-Zone-2			
5.5	59.26	60.19	+ 1.6
4.5	56.55	57.27	+ 1.3
3.5	53.67	53.88	+ 0.4
2.5	50.20	49.80	- 0.8
1.5	46.33	44.69	- 3.6
0.5	39.56	37.86	- 4.5
Blanket-Row-1			
0.5	46.41	42.91	- 8.0
1.5	34.97	34.22	- 2.2
2.5	29.99	28.73	- 4.3
3.5	26.40	25.10	- 5.2
4.5	23.66	22.47	- 5.2
5.5	21.50	20.45	- 5.3

* Normalized to core-peak-power density of 670 KWt/liter

Table 3.20

Gamma-Energy Deposition near the Blanket-Row-3/Reflector
Interface in the 250 MWe Reference
Reactor

Region/Dis- tance from Interface, cm	Gamma Heating Rate, KWt/liter*		Percent Error
	Exact 18-Group Gamma Transport Calculations	Approximate Method Eq. 3.30 & 3.31	
Blanket-Row-3			
5.5	1.82	1.73	- 5.0
4.5	1.72	1.63	- 5.3
3.5	1.65	1.54	- 6.9
2.5	1.59	1.48	- 7.3
1.5	1.56	1.45	- 7.4
0.5	1.45	1.56	+ 7.4
Reflector			
0.5	0.765	0.809	+ 5.5
1.5	0.628	0.661	+ 5.0
2.5	0.563	0.569	+ 1.1
3.5	0.520	0.513	- 1.3
4.5	0.488	0.475	- 2.7
5.5	0.463	0.452	- 2.4
6.5	0.443	0.424	- 4.4
7.5	0.427	0.420	- 1.7

* Normalized to core-peak-power density of 670 KWt/liter.

approximately 8%, a very acceptable figure in view of the 50 to 100% discrepancy between the results using the ORNL and LASL cross section sets, and the relatively large corrections required immediately at the interfaces, where a spike of as much as 100% in the deposition to source ratio may occur.

3.7 Gamma Heating Analysis of Blanket Mockup No. 4

Figure 3.26 shows the schematic arrangement of the Blanket Test Facility (BTF) at the M.I.T. Reactor. Figures 3.27 and 3.28 show the isometric and topviews of Blanket Mockup No. 4. The blanket mockup, driven by a thermal-to-fast neutron converter assembly, is designed to simulate a fast reactor blanket employing UO_2 fuel, stainless steel cladding and sodium coolant. The homogenized atom densities for both Blanket Mockup No. 4 and an equivalent realistic blanket are shown in Table 3.21.

Detailed topical reports are available on the construction of this facility (F-2) and its use for neutronic (L-2) and photonic (S-3) experiments. Thus we will limit discussion here to specific details involving gamma heating.

Figure 3.29 shows the computed total gamma energy source and sink, $\text{Mev/cm}^3\text{-sec}$, in the blanket and the reflector regions of the Blanket Mockup No. 4. The (unshielded) 30-group neutron cross sections that are part

FIG. 3.26 SCHEMATIC CROSS SECTION VIEW OF HOHLRAUM AND BLANKET TEST FACILITY

FIG.3.27 SCHEMATIC VIEW OF BLANKET ASSEMBLY NO. 2

FIG. 3.28 PLAN VIEW OF BLANKET ASSEMBLY SHOWING THE TRAVERSING TUBE POSITIONS

Table 3.21

Homogenized Atom Densities in Blanket No. 4
(Atoms/barn-cm)

Nuclide	Blanket No. 3	Equivalent Realistic Blanket*
U ²³⁵	0.000088	0.000016
U ²³⁸	0.008108	0.008131
O	0.016293	0.016293
Na	0.008128	0.008128
Cr	0.004064	0.003728
Fe	0.013750	0.012611
Ni	0.000000	0.001475
H	0.000073	0.000000
C	0.000096	0.000082
	0.017814	0.017814
Nuclide	Steel Reflector	
C	0.000590	
Fe	0.084570	

* Composed of 37.0 v/o depleted UO₂ (at 90% of the theoretical density), 20.7 v/o Type 316 stainless steel, 32.0 v/o sodium and 10.3 v/o void.

Fig. 3.29 Calculated, Total Gamma Energy Source and Sink in Blanket Mock-up No. 4

of the 42-group LASL coupled neutron-gamma cross section set were used to determine the neutron reaction rates at each spatial point. The gamma energy yields, Mev per reaction, given in Table 3.8 were used. The contribution of inelastic-scattering gamma energy was approximately accounted for by using Fig. 3.10. The method given in Section 3.6 was used to obtain gamma energy deposition rates near the blanket-reflector interface from the computed source.

The gamma dose rates, in rads/hr., were obtained as follows:

$$D_R^H \left(\frac{\text{rads}}{\text{hr}} \right) = D_R^H \left(\frac{\text{Mev}}{\text{cm}^3 \cdot \text{sec}} \right) \cdot \frac{1}{\rho_H} \left(\frac{\text{cm}^3}{\text{g}} \right) \cdot (57.676 \times 10^{-6}) \quad (3.38)$$

where

D_R^H denotes total homogenized gamma dose rate,

ρ_H is the homogenized density,

$\approx 5.61 \text{ g/cm}^3$ for the blanket assemblies,

$\approx 7.87 \text{ g/cm}^3$ for the reflector region, and

57.676×10^{-6} is a conversion factor.

Table 3.22 compares the computed total gamma dose rates with the measured gamma dose rates obtained by Scheinert (S-3) using ^7LiF TLD's. The absolute dose rates

Table 3.22

Comparison of Measured and Calculated Total Homogenized
Gamma Dose Rates in Blanket Mock-up No. 4

Distance from Converter- Blanket Inter- face, cm	Total Homogenized Gamma Dose Rate, Rads/hr		
	Measured	Calculated	
		Absolute*	Relative†
Blanket Assemblies			
5.7	410	386	410
13.2	260	245	259
20.7	200	168	180
28.2	125	114	121
35.7	95	81.5	86.6
43.2	62	61.2	65.0
Reflector			
47.0	25	19.1	20.3
54.6	17	14.2	15.1
62.3	12	9.9	10.5
69.9	9	5.6	6.0
77.5	6	3.1	3.3
85.1	4	1.3	1.4

* Computed values normalized to thermal neutron flux of 3.0×10^9 (#/cm²-sec) in the converter's graphite region.

† Normalized to agree at the first data point in the blanket.

are normalized to a thermal neutron flux of 3.0×10^9 ($\#/cm^2\text{-sec}$) in the converter's graphite region, which was measured by Scheinert (S-3) using a calibrated gold foil. The relative dose rates on the other hand, are normalized to agree at the first data point in the blanket. The agreement between the computed and the measured absolute dose rates is found to be within 20% except in the outer 2 or 3 radial test positions in the reflector region, where both measurements and calculations are less precise.

Table 3.23 compares the computed and measured gamma dose rates in stainless steel. The gamma dose rate in steel, in rads/hr, is obtained as follows:

$$D_R^{S.S.} \left(\frac{\text{rads}}{\text{hr}} \right) = D_R^H \left(\frac{\text{Mev}}{\text{cm}^3\text{-sec}} \right) \cdot \frac{\overline{E\sigma}_a^{S.S.}}{\overline{E\Sigma}_a^H} \cdot N^{S.S.} \left(\frac{1}{g} \right) \cdot (57.676 \times 10^{-6}) \quad (3.39)$$

where

D_R^H is the total homogenized gamma dose rate, Mev/ $\text{cm}^3\text{-sec}$, as given in Fig. 3.29,

$\overline{E\sigma}_a^{S.S.}$ is the microscopic gamma energy absorption cross section for stainless steel, see Table 3.17,

$\overline{E\Sigma}_a^H$ is the macroscopic homogenized gamma energy absorption cross section, obtained from Table 3.17, and

$N^{S.S.}$ is the number of atoms per gram in stainless steel, in units of 10^{24} , ~ 0.01078 per gram.

Table 3.23

Comparison of Measured and Calculated Gamma Dose Rates in
Stainless Steel in Blanket Mock-up No. 4

Distance from Converter- Blanket Inter- face, cm	Gamma Dose Rate in Stainless Steel, Rads/hr		
	Measured	Calculated	
		Absolute*	Relative†
Blanket Assemblies:			
5.7	310	230	310
13.2	210	145	197
20.7	143	100	136
28.2	95	68.3	93
35.7	70	47.6	64
43.2	42	36.6	50
Reflector:			
47.0	25	19.1	24.5
54.6	17	14.2	19.2
62.3	12	10.0	13.5
69.9	9	5.6	7.6
77.5	6	3.1	4.2
85.1	4	1.3	1.8

* Computed values normalized to thermal neutron flux of 3.0×10^9 (#/cm²-sec) in the converter's graphite region.

† Normalized to agree at the first data point in the blanket.

It is seen that better agreement between the computed and measured dose rates is obtained if relative dose rates are compared. Thus part of the observed discrepancies could be attributed to the normalization. It should be recognized that the absolute normalization using the thermal neutron flux in the converter's graphite region is an extremely severe test for the neutronic calculations. In addition the neutronic cross sections were not properly self-shielded (although the spatial shapes of the calculated and measured U-238 capture traverses are in relatively good agreement). Thus, most of the discrepancy may well be attributed to the neutronics rather than the photonics. The relative dose traverses, on the other hand, agree within the experimental precision of about $\pm 10\%$ for the most part.

If the coupled LASL or ORNL cross section sets are used to calculate the gamma dose rates rather than the hybrid method used here (which was given preference because it assures an energy balance), the calculated relative dose rates, as can be seen in Table 3.24, deviate from the experimental dose rates by up to 30% in the blanket region and by as much as a factor of 2 in the reflector region.

Table 3.24

Comparison of Measured and Calculated Total Homogenized
Gamma Dose Rates in Blanket Mockup No. 4

Distance from Con- verter- Blanket In- terface, cm	Total Homogenized Gamma Dose Rate, Rads/hr			
	Measured	Calculated Relative ^a Dose Rate		
		Hybrid Method Assuring Energy- balance ^b	LASL Coupled Cross Section set ^c	ORNL Coupled Cross Sec- tion set ^d
Blanket Assemblies				
5.7	410	410	410	410
13.2	260	259	291	275
20.7	200	180	214	181
28.2	125	121	154	116
35.7	95	86.6	110	77
43.2	62	65.0	81.2	51.6
Reflector				
47.0	25	20.3	47.8	18.8
54.6	17	15.1	32.7	11.6
62.3	12	10.5	20.7	8.6
69.9	9	6.0	12.3	6.0
77.5	6	3.3	6.7	3.6
85.1	4	1.4	2.9	1.7

^aNormalized to agree at the first data point in the blanket.

^bNormalization factor = 1.06

^cNormalization factor = 1.00

^dNormalization factor = 1.01

3.8 The Importance of Bremsstrahlung in the Reflector Region of the 250 MWe Reference Reactor

In the steel reflector region of the FBR, almost all the gammas are produced from captures in Fe, Cr and Ni. More than ~50% of gammas produced from capture events in these materials have energies between 4 to 10 Mev (K-1). Hence the gamma source spectrum is harder than in the core or blanket region. These high energy gammas can give rise to a large number of annihilation gammas, arising from electron-positron pairs produced in pair production events. Furthermore, the electrons and positrons created in the pair-production events, and the electrons resulting from Compton scattering events at high gamma energies, are energetic enough that a significant fraction of their total kinetic energy can be re-emitted in the form of bremsstrahlung radiation, leading to enhanced migration of energy because gammas are more penetrating than electrons.

As indicated earlier in Chapter 1 (Section 1.6), in the coupled neutron-gamma cross section sets, the annihilation gammas are appropriately accounted for as gamma down-scatter events: i.e., the total energy available from these events is accounted for. However, the usually accepted practice of simplifying the problem by not including the effects of bremsstrahlung on energy transport is adopted. In this section, we will assess the validity of this assumption for gamma heating calculations in the

250 MWe reference reactor (see Fig. 3.4).

The bremsstrahlung is taken into account using the following assumptions:

1. The total energy radiated by an electron or positron of kinetic energy T (Mev) moving in a material of atomic number Z is $0.0007 Z T^2$ (Mev), where 0.0007 is a constant whose dimensions are (1/Mev) (see Ref. E-2, p. 615-617).

2. All energy radiated is given off as a single photon at the beginning of the particle path.

The single photon assumption reduces to manageable proportions the number of photons to be kept track of, while accounting for all bremsstrahlung energy. The use of this assumption in the calculation of the contribution of bremsstrahlung to gamma-ray buildup factors results in good agreement with experiments (K-4).

The total bremsstrahlung gamma energy source is then given by:

$$E_B^Y(r) = \sum_{k,g} 2 \cdot \phi_g^Y(r) \cdot \Sigma_{pp,g}^k \cdot C \cdot Z^k \cdot \left(\frac{E_g - 1.02}{2} \right)^2$$

$$+ \sum_{k,g,g'} \phi_g^Y(r) \Sigma_{g \rightarrow g'}^k \cdot C \cdot Z^k \cdot (E_g - E_{g'})^2$$

$$+ \sum_{k,g} \phi_g^Y(r) \cdot \Sigma_{pe,g}^k \cdot C \cdot Z^k \cdot E_g^2, \quad (3.40)$$

where

$E_B^Y(r)$ is the total bremsstrahlung gamma energy source, Mev/cm³-sec, at point r in the reflector,

$\phi_g^Y(r)$ is the gamma flux, photons/cm²-sec, in group g , at point r ,

$\Sigma_{pp,g}^k$ is the macroscopic pair production cross section, cm⁻¹, of material k , group g ,

$\Sigma_{g \rightarrow g'}^k$ is the macroscopic Compton scattering cross section, cm⁻¹, of material k , from gamma group g to g' ,

$\Sigma_{pe,g}^k$ is the macroscopic cross section, cm⁻¹, for the photoelectric effect in material k , group g ,

$C \approx 0.0007$ (1/Mev),

Z^k is the atomic number of material k , and

E_g is the mean energy, Mev, in gamma group g .

Table 3.25 summarizes the results obtained using the 42-group LASL coupled neutron-gamma cross section set. It is seen that the bremsstrahlung gamma energy source is of the order of ~5% of the primary neutron-induced gamma energy source, at almost all points in the reflector.

In view of the results obtained in Section 3.6, where it is shown that, except for the leakage of gammas from the blanket to the reflector, the migration of even the

Table 3.25

Bremsstrahlung Gamma Energy Source in the Reflector Region
of the 250 MWe Reference Reactor

Distance from Blanket- Reflector In- terface, cm	Primary (Neutron-induced) Gamma Energy Source KWT/ liter*	Secondary Bremsstrahlung Gamma Energy Source, KWT/liter	Ratio of Bremsstrah- lung to Primary Gamma Energy Source, Per- cent
2.5	0.473	0.0215	4.5
7.5	0.375	0.0201	5.3
12.5	0.279	0.0148	5.3
17.5	0.200	0.0111	5.5
22.5	0.140	0.0077	5.5
27.5	0.097	0.0054	5.6
32.5	0.065	0.0036	5.6
37.5	0.041	0.0023	5.6
42.5	0.022	0.0013	5.9
47.5	0.008	0.0005	6.3

* Normalized to core-peak-power density of 670 MWt/liter.

primary (neutron-induced) gammas inside the reflector region does not affect gamma heating rates (i.e. the local source/local deposition ratio ≈ 1.00), it can be concluded that the migration of the secondary bremsstrahlung gamma energy source, which is a small fraction ($\sim 5\%$) of the primary gamma energy source, and which (being at lower energy) has a shorter mean free path than the primary radiation, will not alter the distribution of gamma heating rates in the reflector. Because of the softer gamma source spectrum, the relative contribution and resulting effects will be even smaller in the core and blanket. Thus the results obtained here indicate that for gamma heating calculations in the reflector region of the FBR, the approximation of neglecting the effects of bremsstrahlung radiation is justified. Again note that the present methods correctly account for all energy deposited: the only possible effect of bremsstrahlung could be on its spatial distribution.

3.9 Summary and Conclusions

In this chapter, a sequence of topics relevant to gamma heating analyses in homogenized fast reactor media have been considered. In the introductory Section, 3.1, the various quantities of interest in gamma heating calculations were examined and associated uncertainties were assessed. In Section 3.2, the reference reactor configurations and compositions used for gamma heating analyses in this chapter were described.

In Section 3.3 it was shown that the uncertainties in gamma production spectra, which are presently of the order of 100% or more, are largely mitigated in gamma heating calculations if the total amount of gamma energy produced in any event is accurately preserved--a step which is apparently omitted in preparation of current coupled neutron-gamma cross section sets. In Section 3.4 it was shown, using the 26-group (ABBN) neutron cross section set (B-2), that ~90% of the gamma energy source in the harder core neutron spectra, and more than 95% of the gamma energy source in the softer blanket and reflector neutron spectra is accurately calculated by ignoring the effect of neutron energy in all capture and fission reactions--i.e. using thermal neutron energy yields. It was also shown that the remaining 10% or so of the gamma energy source in the core region (less than 5% in the

blanket and reflector regions) comes mainly from inelastic scattering.

In Section 3.5, the gamma heating analyses for the 250 MWe reference reactor (see Fig. 3.4) and for a similar 3000 MWe reactor system (see Fig. 3.5), done using coupled neutron-gamma cross section sets, were compared and evaluated. It was found that on the whole there are no profound differences in the behavior of gammas in the 250 MWe and 3000 MWe designs. It was shown that at the present stage of coupled neutron-gamma cross section set development, the gamma energy source as calculated by the coupled sets is unreliable for fast reactor gamma heating analysis, due to inconsistencies in gamma production cross section data. Thus it is concluded that the gamma energy source in fast reactor media is, at present, more reliably and accurately estimated by explicitly utilizing gamma energy yields per fission and per capture in each material. If all capture and fission gamma energy yields are assumed to be thermal, the overall error in the contribution from these sources is found to be $\sim 1\%$ (see Section 3.4.2). Neglecting inelastic-scattering gammas, however, could introduce as much as 10% error in the core region and less than 5% in the blanket and reflector regions. The inelastic-scattering gamma energy, as estimated by neutron energy balance considerations, is found

to be roughly comparable to neutron heating by (primarily) elastic scattering reactions (see Fig. 3.11).

In Section 3.6, the leakage of gammas from the core region to the blanket, and from the blanket to the reflector region was examined. It was found that the leakage gamma energy is deposited within the first few transport-mean-free-paths of the boundary. Averaged over the entire radial blanket region of the reactor, the leakage of gammas from the core to the blanket accounts for less than $\sim 2\%$ of the total blanket gamma heating; averaged over the entire reflector region (50 cm here), the leakage of gammas from the blanket to the reflector accounts for not more than $\sim 4\%$ of the total reflector gamma heating. However, as mentioned above, the leakage gamma energy is deposited near the boundaries, and can increase local gamma heating near the boundary by as much as a factor of 2. A simple method based on imposition of a gamma energy balance across the interfaces was developed to calculate the distribution of leakage gamma energy near the boundary, and was found to agree with "exact" transport calculations within a maximum discrepancy of 8%.

In Section 3.7, a gamma heating analysis of Blanket Mock-up No. 4 irradiated in the Blanket Test Facility (BTF) at the M.I.T. Reactor was presented. It was found that if the dose rates are calculated by explicitly utilizing the gamma energy yields per reaction, the calculated

relative dose rates agree with the measured dose rates within the experimental precision of $\pm 10\%$, except in the outer 2 or 3 test positions in the reflector region, where both measurements and calculations are less precise. On the other hand, if the LASL and the ORNL coupled neutron-gamma cross section sets are used, the calculated relative dose rates deviate from the experimental dose rates by up to 30% in the blanket region and by as much as a factor of 2 in the reflector region.

Finally, in Section 3.8, the phenomenon of bremsstrahlung is shown to be negligible in fast reactor core, blanket and reflector media.

Chapter 4

SUMMARY, CONCLUSIONS AND RECOMMENDATIONS

4.1 Introduction

Specification of the local energy deposition rate density is a key link between the neutronic and thermal-hydraulic design for any nuclear reactor. The gamma (as opposed to fission or neutron) heating component is of particular interest in the blanket (reflector) region of fast breeder reactors, where it is a major (dominant) fraction of the total heating rate. The primary objectives of the work reported here have been to conduct an in-depth examination of the gamma heating phenomenon in FBR media, evaluate state-of-the-art methods for its computation, test common simplifying assumptions such as those allowed by the draft ANSI standard "N676 D-2" (A-1) develop improved approaches where appropriate, and, finally to evaluate the experimental measurements carried out previously by Scheinert on an LMFBR blanket mockup at M.I.T. (S-3).

4.2 The Nature of Gamma Heating

Energy is generated in the fast breeder reactor when a particle, usually a neutron, (and to a much lesser extent gammas, protons, etc.) initiates a reaction in any of the reactor's constituent materials that leads to net energy-release. If the minor reactions $(n, 2n)$, (n, p) , (n, α) , (γ, n) , (γ, f) , etc. are neglected, as was done in

this study, the only two primary sources of energy in the reactor are neutron fission and neutron capture. These two reactions, and the various particles produced in these reactions, lead to the following five primary sources of gamma-energy in the reactor:

1. Prompt and decay gammas from neutron fission.
2. Prompt and decay gammas from neutron capture.
3. Gammas from inelastic scattering of neutrons emitted in the fission process.
4. Gammas arising from the annihilation of positrons (β^+) emitted in the fission and capture reactions (rare--hence negligible--because fission and capture products are neutron-rich and generally undergo β^- decay).
5. Bremsstrahlung gammas arising from the slowing down of beta particles emitted in the fission and capture reactions: negligible in view of the low mean energy of these particles.

These primary gammas deposit their energy by pair production, Compton scattering and the photoelectric effect; these three reactions cover the entire range of important gamma interactions of interest in fast reactor gamma heating analyses (E-2).

The above mechanisms of primary gamma energy deposition lead in turn to the following two secondary sources of gammas in the reactor: annihilation gammas produced

subsequent to the pair production events; and bremsstrahlung gammas arising from the slowing down of electrons and positrons created in the pair production, Compton scattering and photoelectric interactions.

Table 4.1 summarizes the various phenomena involved and the extent to which they were investigated in the present work.

4.3 Heterogeneity and Homogenization

A determination of the level of complexity (number of spatial mesh points and energy groups) necessary for accurate gamma heating calculations is an essential step in any computational approach. Thus the first topic examined in depth was whether homogenization could be employed on the unit cell level. Since gamma cross sections (unlike neutron cross sections) do not exhibit sharp resonances, one might expect heterogeneity to be less important in photonic calculations; on the other hand energy-spectrum-averaged gamma transport mean free paths in FBR media are approximately a factor of three shorter than those for neutrons, which requires that a more detailed examination be performed.

A second point in favor of simplification is that unlike the situation in a neutron balance, where detailed knowledge of the history and fate of all neutrons must be

Table 4.1

Summary of Gamma Energy Sources Considered

<u>Gamma Energy Source</u>	<u>Comment</u>
(A) <u>Primary Gamma Energy Sources</u>	
1. Fission Gammas (Prompt and Decay)	Investigated here as an important source
2. Capture Gammas (Prompt and Decay)	Investigated here as an important source.
3. Inelastic Scattering Gammas	Investigated here as an important source; found <u>not</u> negligible
4. Gammas from (n,2n) and (n, charged particle) Reactions	Considered negligible; thresholds for these reactions are high and cross sections low.
5. Annihilation Gammas	Considered negligible; primary positron sources rare in the reactor because fission and capture products are neutron-rich and generally undergo β^- decay.

(continued)

Table 4.1 (Continued)

Summary of Gamma Energy Sources Considered

<u>Gamma Energy Source</u>	<u>Comment</u>
(A) <u>Primary Gamma Energy Sources (continued)</u>	
6. Bremsstrahlung gammas	Considered negligible in view of the low mean energy of betas emitted subsequent to fission and capture reactions.
(B) <u>Secondary Gamma Energy Sources</u>	
1. Annihilation Gammas	Not negligible; secondary sources of positrons subsequent to pair production events significant in all the three regions--core, blanket, reflector--of the reactor; included in gamma transport calculations.
2. Bremsstrahlung Gammas	Investigated; however, its effect on gamma transport was found negligible in all three regions--core, blanket, reflector--of the reactor.

determined with high precision due to the necessity of achieving a chain reaction, gamma interactions are of interest solely with respect to their ultimate contribution to local thermal energy generation, and engineering accuracy will suffice.

To examine heterogeneous effects, attention was focused on two geometrical configurations: a representative three-region cylindricized unit cell for an LMFBR core zone, having a 0.67 cm diameter fuel pin in 0.05 cm thick stainless steel cladding and ~ 40 % of sodium coolant; and a hypothetical multi-cell slab core consisting of alternating 2cm thick layers of fuel (mixed UO_2 and PuO_2) and coolant (Na), steel-reflected at its outer periphery. The gamma energy source and sink calculations at each spatial point, for each material, were done using the ORNL 40-group (0-1) and the LASL 42-group (S-4) coupled P_3 neutron-gamma cross section libraries, and a one-dimensional discrete ordinate transport code, ANISN (E-1), in the S_8P_3 approximation, and also, where found adequate, in the S_8P_1 approximation.

A careful analysis of the results obtained, as summarized in Table 4.2, indicates that a volume-homogenized representation of fast reactor media is entirely satisfactory for gamma energy production and deposition calculations. The effect of heterogeneity, in general,

Table 4.2

Comparison of Gamma Heating Calculations for Heterogeneous
and Volume-Homogenized Unit Cells

Quantity of Interest, Mev/cm-sec*	Gamma Energy Source and Sink Calculated Using ORNL Coupled Cross Section Set		Ratio of ORNL σ -set Results to LASL σ -set Results
	Homogeneous Unit Cell	Heterogeneous Correction factor**	
Gamma Energy Source			
Total	9.833	1.0001	1.56
Fuel	9.329	1.0001	1.63
Clad	0.377	1.0000	0.73
Coolant	0.127	1.0000	1.59
Gamma Energy Sink			
Total	9.745	1.0012	1.52
Fuel	8.492	1.0014	1.53
Clad	0.966	0.9990	1.50
Coolant	0.287	0.9965	1.54

* Normalized to a linear power density of 1 fission neutron/cm-sec.

** Ratio of heterogeneous to homogeneous results.

is to enhance the gamma energy deposition in the fuel region by 1 to 2 parts in 1000. For practical purposes, this should be compared with the basic uncertainty of 30 to 50 parts in 1000, in the neutron-induced gamma energy yields. This additional energy deposited in the fuel region if a heterogeneous representation is used is of the order of 0.01 Mev per fission neutron. This should be compared to a total of 65 to 70 Mev deposited in the fuel region per fission neutron. Its effect on fuel centerline temperature, and other related parameters of engineering interest is therefore negligible. Note, in Table 4.2, that the overall difference in gamma energy yields between the two coupled cross section sets is roughly 60%. This surprisingly large discrepancy will be discussed further in Section 4.4.4.

The effect of heterogeneity on gamma energy transport was found to be similarly negligible, particularly in view of the estimated uncertainty of $\pm 10\%$ in gamma transport cross sections (K-1). The maximum difference in gamma dose rates between the homogeneous and heterogeneous cases for the steel reflector region of the hypothetical slab core was found to be $\sim 0.2\%$, far below these more basic uncertainties.

A final point investigated involved the sharp K-edge of the photoelectric absorption cross section, which

is superficially similar to the sharp edge of a neutron absorption resonance. Calculations showed that even an (arbitrary) order of magnitude reduction (i.e. heterogeneous shielding factor of 0.1) in the gamma group cross sections in this region had a negligible effect (~ 1 part in 800) on local gamma heating; hence this phenomenon was judged negligible without a more detailed analytic investigation.

Based on the preceding results it was concluded that a volume-homogenized representation of fast reactor lattices is adequate for the analysis of gamma energy production, transport and deposition in FBR media.

4.4 Gamma Heating Analyses in Homogenized Fast Reactor Media

A sequence of topics bearing on the amount and quality of detail which must go into gamma heating analyses in homogenized fast reactor media have been considered during the course of this work. Inconsistencies in state-of-the-art formulations have been brought out which have previously been masked by a certain degree of over-sophistication, and some simple approaches which satisfy a gamma energy balance have been introduced.

4.4.1 Reference Reactor Configurations Used for Gamma Heating Analyses

For most of the gamma heating analyses in this work, a radial one-dimensional version of the 250 MWe LMFBR configuration, shown in Fig. 4.1, was employed. The main features of this reactor are: the two approximately equal-volume, oxide-fueled core zones (30 % at 85% t.d. mixed uranium and plutonium oxide, 50 % of sodium coolant and 20 % of stainless steel structure); three oxide-fueled blanket rows (50 % at 95% t.d. uranium oxide, 30 % sodium coolant and 20 % stainless steel structure), each 15 cm thick; a 50 cm thick stainless steel radial reflector with 10 % of sodium coolant; core averaged power density ~ 450 KWt/liter (core peak power density ~ 670 KWt/liter) and an assumed system thermal efficiency of 40%. To investigate the effect of reactor size on gamma heating analyses, an LMFBR design at the other extreme of interest--a large 3000 MWe oxide-fueled reactor, with six core zones, three blanket rows and a stainless steel reflector was used. Tagishi has extensively studied the breeding neutronics of these designs (T-2).

4.4.2 The Effect of Gamma-Production Spectra on Gamma Heating Analyses

The uncertainty in gamma-production spectra can be large. Even in a coarse gamma group structure (5 to 10),

Fig. 4.1 Schematic Elevation View of the Upper Half of the 250 MWe Reference Reactor

the uncertainty in a particular group can be as large as 50% or even more (B-3, T-1). Further, data on neutron-energy-dependent gamma production spectra from all reactions and for all materials of interest is lacking. It is of considerable interest, therefore, to see the effect of production spectrum shape on gamma heating calculations. For this purpose, two arbitrarily chosen gamma-production spectra shown in Fig. 4.2, were introduced in the ORNL coupled neutron-gamma cross section set, without affecting the total gamma energy yield from each neutron group. The same spectra were used for all neutron groups and for all materials (i.e. oxygen, sodium, iron, U-235, U-238, Pu-239 and Pu-240) needed for these calculations. By comparing reactor calculations using the two gamma cross section sets generated in this manner, it was found that uncertainties in gamma production spectra are largely mitigated in gamma heating calculations if the total amount of gamma energy produced in any event is accurately preserved--a step which is apparently omitted in the preparation of current coupled neutron-gamma cross section sets (see Section 4.4.4.). The maximum difference in gamma heating rates was found to be an insignificant $\pm 4\%$ near the core-blanket and the blanket-reflector interfaces, where a spike of as much as 100% in the deposition to source ratio can occur. Note, furthermore, that the

Fig. 4.2 Unnormalized Gamma-Source-Spectra
Used to Study the Sensitivity of Gamma
Heating Calculations to Gamma-Source-
Spectra

two hypothetical production spectra of Fig. 4.2 provide a very strict test of the sensitivity of gamma heating calculations to the gamma production spectra. Spectrum-1 in Fig. 4.2 introduces gammas mainly in the middle of the gamma energy range, and spectrum-2 introduces almost all gammas at very low or very high energies. Thus it can be concluded that the uncertainties in gamma production spectra are not likely to be a significant source of error in gamma heating analyses of FBR Media.

4.4.3 Contribution of Neutron-Energy to Gamma Energy Source

Fission neutrons are born in the reactor at an average energy of ~ 2 Mev. A part of this energy is deposited through recoils on nuclei, primarily via elastic scattering collisions. Most of the rest appears in the form of gamma energy as the nuclei lose their energy of excitation subsequent to inelastic scattering collisions. The neutron energy also affects gamma energy yields in capture and fission reactions. If E_c^{th} and E_f^{th} are the gamma energy yields per capture and per fission, respectively, at thermal neutron energy, the gamma energy yields, E_c^g and E_f^g , in these reactions at a higher neutron energy, E_g , in the neutron group g , can be estimated as follows (K-1):

$$E_c^g = E_c^{th} + \frac{A}{A+1} E_g, \text{ and} \quad (4.1)$$

$$E_f^g \approx \left(\frac{E_g + T}{T} \right) E_f^{th}, \quad (4.2)$$

where A is the mass number of the material and T is the total energy released in fission (~ 207 Mev). The gamma energy yield, $E_{\gamma, g \rightarrow g'}^{in}$, in an inelastic collision in which the neutron transfers from neutron group g to g' is given by:

$$E_{\gamma, g \rightarrow g'}^{in} = \frac{A}{A+1} E_g - \frac{A-1}{A} E_{g'}. \quad (4.3)$$

Figure 4.3 compares the inelastic-scattering gamma-energy source to the total gamma-energy source in the 250 MWe reference reactor, calculated using the 26-group Russian (also known as ABBN or Bondarenko) cross section set (B-2). It can be seen that the inelastic-scattering gamma energy can be significant, especially in the core region, in blanket-row-1 and in the reflector, near the blanket-reflector interface. A comparison of inelastic-scattering gamma energy with neutron heating via elastic recoils on the nuclei, calculated using the same

Fig. 4.3 Contribution of Inelastic-Scattering Gammas to Total Gamma-Energy Source in the 250 MWe Reference Reactor

cross section set, shows that in the core region inelastic (gamma) heating is 60 to 70% larger than the elastic recoil heating; in the blanket region the neutron energy lost in elastic and inelastic collisions is roughly comparable; and in the reflector region comparatively more neutron energy is lost in elastic collisions.

The effect of neutron-energy on the gamma energy source from capture and fission reactions calculated using Eqs. 4.1 and 4.2, is, however, found to be comparatively negligible (less than 1%). Thus it can be concluded that the use of thermal-neutron-induced gamma energy yields for all captures and fissions introduces less error in the determination of gamma energy sources than neglecting the inelastic-scattering gammas (as is more often done; see for example Ref. K-1).

4.4.4 Gamma Heating Analyses Using ORNL and LASL Coupled Neutron-Gamma Cross Section Sets

Figure 4.4 compares the gamma heating rates for the 3000 MWe reactor, calculated using the ORNL 40-group (O-1) and the LASL 42-group (S-4) coupled neutron-gamma cross section sets, normalized to the same core peak power density of 700 KWt/liter. Figure 4.5 compares the ratios of the gamma heating rate to the local fission heating rate.

Fig. 4.4 Gamma Heating Rate in a 3000 MWe Reactor

Fig. 4.5 Ratio of Gamma Heating Rate to Local Fission Power in 3000 MWe Reactor

It can be seen that the difference in gamma heating rates obtained from the two coupled cross section sets is ~50 to 70%. In blanket-row-1, gamma heating is comparable to fission heating, and in the outer rows of the blanket, gamma heating exceeds fission heating (in blanket-row-3, gamma heating is ~2.5 times the fission heating, under beginning-of-life conditions, prior to buildup of bred plutonium). In the reflector external to the blanket gamma heating constitutes more than 95% of the total heating. The gamma heating rates in the smaller 250 MWe reactor, calculated using the same coupled neutron-gamma cross section sets, were found to be quite similar to those in the larger 3000 MWe design. Hence there are no important effects associated with reactor size--an important conclusion with respect to the validity of computational methods and the use of demonstration size reactors to test design methods.

Figure 4.6 compares the gamma energy yields in the two coupled cross section sets for U-238, in neutron spectra appropriate to metal, carbide and oxide fueled fast reactor cores and blankets. In this comparison, inelastic scattering gamma energy is neglected, i.e., the approximation,

Fig. 4.6 Plot of Equation 4.4 for U-238

$$E_Y \approx E_C \cdot C + E_f \cdot F, \quad (4.4)$$

is used, where

E_Y is the total gamma energy yield, Mev,

C is the number of captures,

F is the number of fissions, and

E_C and E_f are constants.

It can be seen that the difference in gamma energy yields for U-238 can be as large as 40% between the two coupled cross section libraries.

However, a more appropriate (and exact) check is made by examining gamma energy yields from individual neutron groups. Table 4.3 gives the gamma energy yields from neutron groups 15 to 22 in the ORNL coupled cross section set for U-238. (The older version of this cross section library, obtained in 1972, was used for most of the work in this report. As noted in Table 4.3, the gamma energy yields for U-238 in the new version of this library, acquired in June 1975, are same as in the older version). The upper energy bound of the neutron groups examined here is below inelastic and fission thresholds. Thus radiative capture of neutrons is the only gamma-producing reaction in these groups. The gamma-energy yields per capture, $E_{Y,C}^i$ (Mev), for these neutron groups can there-

Table 4.3

Neutron-Induced Gamma Energy Yields from U-238 in the ORNL
Coupled Neutron-Gamma Cross Section Set

Neutron Group g	Neutron Energy Range in Group g ^a	Gamma Energy Yield per Capture in Group g, Mev	
		Old Version ^b	New Version (June 1975)
15	3.35-0.583 kev	7.062 ^c	Same as in Old Version
16	0.583-0.101 kev	4.959	
17	101-29 ev	7.320	
18	29-10.7 ev	9.685	
19	10.7-3.06 ev	14.223	
20	3.06-1.12 ev	4.473	
21	1.12-0.414 ev	4.480	
22	Thermal	4.489	

^aNeutron energies in these groups are below inelastic and fission thresholds; thus capture is the only gamma-producing reaction. Any significant variation of gamma energy yield per capture in these groups is essentially the uncertainty in gamma production cross section data, a factor of 3 here, which affects gamma energy yield by the same factor; actual uncertainty in gamma energy yield is probably 5 to 10%.

^bThis version was used for most of the work in this report.

^cA reasonable value of gamma energy yield per capture in all these groups would be ~4.8 Mev (K-1).

fore be calculated as follows:

$$E_{\gamma,c}^i = \sum_{j=1}^G \phi_i \cdot \sigma_{i \rightarrow j}^k \cdot N^k \cdot E_j / \phi_i \cdot \sigma_{c,i}^k \cdot N^k, \quad (4.5)$$

where

i denotes neutron group,

j denotes gamma group,

k denotes the material (U-238 here),

ϕ denotes neutron group flux, $\#/\text{cm}^2\text{-sec}$,

$\sigma_{i \rightarrow j}^k$ is the gamma production cross section,

$\sigma_{c,i}^k$ is the neutron capture cross section,

E_j is the mean energy of gamma group j , and

G is the total number of gamma groups.

It is seen that the gamma energy yields per radiative capture in these neutron groups vary by as much as a factor of 3, while the actual uncertainty in the gamma energy yield per radiative capture in U-238 is $\sim \pm 5\%$ or less.

Table 4.4 gives gamma energy yields per capture in Fe and Na calculated for neutron groups 16 to 23 in the LASL coupled cross section set. Again, the radiative capture is the only gamma-producing reaction in these neutron groups. The variation in gamma energy yields is

Table 4.4

Neutron-Induced Gamma Energy Yields from Fe and Na in the
LASL Coupled Neutron-Gamma Cross
Section Set

Neutron Group g	Gamma Energy Yield per Capture in Group g, Mev	
	Fe	Na
16 ^a	7.65 ^b	1.03 ^c
17	7.90	0.73
18	7.94	13.30
19	76.13	3.55
20	360.30	4.04
21	8.21	3.98
22	6.86	4.33
23	7.58	5.22

^a Neutron energies in these groups are below inelastic threshold; thus capture is the only gamma-producing reaction.

Any significant variation of gamma energy yield per capture in these groups is essentially the uncertainty in gamma production cross section data or an error.

^b A reasonable value of gamma energy yield per capture in all these groups for Fe would be ~7.8 Mev (K-1).

^c A reasonable value of gamma energy yield per capture in all these groups would be ~7 Mev, excluding post-capture decay gammas (K-1).

found to be as much as a factor of 10, or even more.

Similar inconsistencies in gamma energy yields are found in other materials needed for gamma heating analyses in fast reactor media. Such inconsistencies make gamma heating calculations done using the coupled neutron-gamma cross section libraries unreliable and inaccurate. Thus it is concluded that, at present, the gamma-energy source in fast reactor media is more reliably and accurately estimated by explicitly emphasizing an energy balance in all neutron-induced gamma-producing reactions.

If there is one favorable aspect in the present status of the ORNL and LASL coupled neutron-gamma cross section sets, it is that they conservatively overestimate gamma heating in the blanket and reflector regions (when normalized to the correct local neutron capture rate). Hence thermal/hydraulic designs based on their use will lead to local over cooling rather than over heating. In the core region the LASL set underestimates gamma heating by approximately 20%; however, here gamma heating is a relatively less important contribution to the total.

4.4.5 Leakage of Gamma Energy Across Interfaces

Figure 4.7 shows the local gamma-energy deposition (gamma heating)-to-source ratio in the 250 MWe reference reactor. Although gamma-leakage from the core region to the blanket accounts for less than ~2% of the blanket gamma

Fig. 4.7 Local Gamma-Energy Deposition-to-Source Ratio in the 250 MWe Reference Reactor

heating, and the gamma-leakage from the blanket region to the reflector accounts for not more than ~4% of the total reflector gamma heating, the leakage gamma-energy is deposited within a few transport-mean-free-paths of the boundaries, and a spike in the gamma-energy deposition-to-source ratio as large as 100% can occur (see Fig. 4.7). It was found that an approximate method imposing an energy balance across the interfaces, together with the continuity of gamma flux at the interfaces and the use of the gamma-spectrum-weighted transport-mean-free-path as the exponential relaxation length for gamma-energy deposition near the interfaces, calculates these spikes within a maximum discrepancy of 8% (see Table 4.5), a very acceptable figure in view of the 50 to 100% discrepancy in gamma heating rates if the coupled neutron-gamma cross section sets are used, and the relatively large corrections required immediately at the interfaces. The success of this model and the previously noted insensitivity to gamma source spectrum also provide strong circumstantial evidence that only one energy group may suffice for most gamma heating calculations in FBR media.

4.4.6 The Effect of Bremsstrahlung on Spatial Redistribution of Gamma Energy Deposition in Fast Reactor Media

An analysis of the bremsstrahlung phenomenon in the steel reflector region of the 250 MWe reference reactor, where the gammas, and, therefore, the resulting betas in

Table 4.5

Gamma-Energy Deposition near the Blanket-Row-3/ Reflector
Interface in the 250 MWe Reference
Reactor

Region/Dis- tance from Interface, cm	Gamma Heating Rate, Kwt/liter*		Percent Error
	Exact 18-Group Gamma Trans- port Calcula- tions	Approximate Method	
Blanket-Row-3			
5.5	1.82	1.73	-5.0
4.5	1.72	1.63	-5.3
3.5	1.65	1.54	-6.9
2.5	1.59	1.48	-7.3
1.5	1.56	1.45	-7.4
0.5	1.45	1.56	+7.4
Reflector			
0.5	0.765	0.809	+5.5
1.5	0.628	0.661	+5.0
2.5	0.563	0.569	+1.1
3.5	0.520	0.513	-1.3
4.5	0.488	0.475	-2.7
5.5	0.463	0.452	-2.4
6.5	0.443	0.424	-4.4
7.5	0.427	0.420	-1.7

* Normalized to core-peak-power density of 670 kwt/liter.

pair production, Compton scattering and photoelectric interactions, are particularly energetic, shows that the total bremsstrahlung energy re-radiated during the slowing down of the betas is between 4 to 6% in this region. In view of the fact that the gamma-energy deposition-to-source ratio is essentially 1.00 for even the primary neutron-induced gammas in most of the region and the relatively shorter mean-free-path of the softer bremsstrahlung radiation, it is concluded that the effect on spatial re-distribution of gamma energy deposition due to the enhanced migration of bremsstrahlung radiation can be neglected. Because of the lower energies of betas in the core and the blanket, the resulting bremsstrahlung-related effects will be even smaller in these regions.

4.4.7 Gamma Heating Analysis of Blanket Mockup No.4

Figure 4.8 shows the calculated gamma-energy source and sink, $\text{Mev/cm}^3\text{-sec}$, in the blanket and reflector regions of Blanket Mockup No. 4 (a representative UO_2 fueled LMFBR blanket similar to the radial blanket of the reactor shown in Fig. 4.1), irradiated in the Blanket Test Facility (BTF) at the M.I.T. Reactor (F-2). In view of the inconsistencies in gamma energy yields in the coupled neutron-gamma cross section sets (see Section 4.4.4), a hybrid method assuring energy balance for each gamma-producing neutron reaction was used. Contributions from capture,

Fig. 4.8 Calculated Total Gamma Energy Source and Sink in Blanket Mock-up No. 4.

fission and inelastic-scattering reactions were included in the calculation of the gamma energy source shown in Fig. 4.8. The gamma-energy sink was then determined by imposing a gamma energy balance across the blanket-reflector interface, as outlined in Section 4.4.5.

Table 4.6 compares the computed total homogenized gamma dose rates (rads/hr) with the measured gamma dose rates obtained by Scheinert using ^7LiF TLD's (S-3). The dose rates obtained using the LASL and the ORNL coupled neutron-gamma cross section sets are also included for comparison. It is seen that if the dose rates are calculated by explicitly utilizing the gamma energy yields per reaction (which assures an energy balance), the calculated relative dose rates agree with the measured dose rates within the experimental precision of $\pm 10\%$, except in the outer 2 or 3 test positions in the reflector region, where both measurements and calculations are less precise. On the other hand, if the LASL and the ORNL coupled neutron-gamma cross section sets are used, the calculated relative dose rates deviate from the experimental dose rates by up to 30% in the blanket region and by as much as a factor of 2 in the reflector region.

4.5 Recommendations for Future Work

It has been found during the course of this work that in present coupled neutron-gamma cross section sets, there

Table 4.6

Comparison of Measured and Calculated Total Homogenized
Gamma Dose Rates in Blanket Mockup No. 4

Distance from Converter- Blanket Interface, cm	Total Homogenized Gamma Dose Rate, Rads/hr			
	Measured	Calculated Relative ^a Dose Rate		
		Hybrid Method Assuring Energy- Balance ^b	LASL Coupled Cross Section Set ^c	ORNL Coupled Cross Sec- tion Set ^d
Blanket Assemblies				
5.7	410	410	410	410
13.2	260	259	291	275
20.7	200	180	214	181
28.2	125	121	154	116
35.7	95	86.6	110	77
43.2	62	65.0	81.2	51.6
Reflector				
47.0	25	20.3	47.8	18.8
54.6	17	15.1	32.7	11.6
62.3	12	10.5	20.7	8.6
69.9	9	6.0	12.3	6.0
77.5	6	3.3	6.7	3.6
85.1	4	1.4	2.9	1.7

^a Normalized to agree at the first data point in the blanket

^b Normalization factor = 1.06

^c Normalization factor = 1.00

^d Normalization factor = 1.01

3

is lack of energy balance in the neutron-induced gamma-energy yields. This directly affects the determination of the gamma energy source--the single most important quantity that enters into any analysis of gamma heating (indeed local source may be equated to local deposition in most of the core, blanket or reflector). The actual uncertainties in total gamma energy yields per reaction in most cases need not be greater than $\pm 5\%$ whereas in the present coupled neutron-gamma cross section sets, energy yields vary by as much as a factor of 3 or even more from their theoretical limits. This makes gamma heating calculations done using coupled neutron-gamma cross section libraries unreliable and inaccurate to a large extent. Thus an energy balance in the neutron-induced gamma energy yields should be systematically checked, and imposed where it is lacking, for each neutron group and each reaction, and for all materials, to make these coupled libraries useful tools for gamma heating analysis. For example, Eq. 4.5 may be employed to check capture reaction gamma energy yields.

The present analysis has focused on gamma heating in large homogeneous media and near the interfaces of such regions. Some additional work should be done on local heterogeneity of small extent, in particular on such items as control poison pins. Another topic worth further inves-

tigation is the number of gamma energy groups required. Our work has suggested that considerably fewer gamma energy groups (1 to 3) may suffice; this would represent a significant economy relative to the 18 gamma groups of the ORNL coupled set and the 12 gamma groups of the LASL coupled set.

Although we have focused on the LMFBR, most of the results obtained in this work also apply directly to the GCFR as well. There is one aspect worth further investigation, both analytically and experimentally: the effect of streaming (anisotropic diffusion) through the essentially transparent coolant passages in the GCFR-- especially in the blanket where source gradients are steep.

The work reported here has covered many aspects treated in ANSI Standard N676. In general our results substantiate many of the allowable approximations under this standard. However, in future drafts of the standard it is recommended that greater distinction be drawn with respect to FBR blankets, where normalization of the gamma energy source solely to fission rate is not as appropriate as it is in the core. The large deviation in gamma deposition-to-source ratios possible near region interfaces should also be noted. The results reported here may also permit authorization of further routine simplifications by the standard should similar results applicable to other

reactor types, LWR's in particular, permit such generalization.

With regard to the experimental work using the Blanket Test Facility (BTF) at the M.I.T. Reactor, an attempt should be made in future mockups to relate the photonics and the neutronics of fast reactor blankets (and reflectors) on an absolute basis by measuring absolute gamma heating to neutron capture ratios. This may be possible through simultaneous irradiation of foils and TLD's in the blanket and in the graphite-lined thermal neutron hohlraum. In the latter facility, which drives the Blanket Test Facility, well-thermalized neutrons and a gamma spectrum derived from carbon capture gammas should be in equilibrium.

Finally, in recapitulation, Table 4.7 lists the major conclusions reached in the subject research.

Major Conclusions Reached: Gamma Heating in FBR's

- * Volume-weighted homogenization is permissible.
- * Gamma-source-spectrum shape is relatively unimportant.
- * Inelastic-scattering de-excitation gammas should be considered explicitly.
- * Use of thermal-neutron-induced gamma energy yields for capture and fission is adequate.
- * There are no important effects associated with reactor size.
- * Energy balances on total gamma energy yield are lacking in current coupled neutron-gamma cross section sets and should be incorporated.
- * Considerably fewer gamma energy groups can be used than in current neutron-gamma cross section sets.
- * Current neutron-gamma cross section sets generally calculate conservatively high gamma heating when normalized to correct neutron capture rates.
- * The energy deposition-to-source ratio is ~ 1.00 inside large homogenized regions (more than ~ 10 cm from interfaces with media which differ greatly in composition or gamma source densities).
- * At interfaces the deposition-to-source ratio can be as high as 2.0.
- * Interface effects can be adequately described using a simple one-gamma-group attenuation kernel method developed in the present work.
- * Bremsstrahlung effects can be neglected.
- * ANSI Standard N676 approximations are generally substantiated here; some reservations exist on applicability to FBR blankets.
- * M.I.T. Blanket Mockup No. 4 gamma heating measurements and our "best-method" calculations are in good agreement.

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