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LMFBR BLANKET PHYSICS PROJECT PROGRESS REPORT NO. 6

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ABSTRACT

This is the sixth annual report of an experimental and analytical program for the investigation of the neutronics and photonics of benchmark mockups of LMFBR blankets.

During the period covered by the report, July 1, 1974 through June 30, 1975, work was devoted primarily to a wide range of analytical/numerical investigations, including blanket fuel management/economics studies, evaluation of improved blanket designs, and assessment of state-of-the-art methods for gamma heating calculations.

Experimental work included preparations for resumption of MIT Reactor operations, primarily fabrication of improved steel reflector assemblies for blanket mockups, and development of an improved radiophotoluminescent readout device for LiF thermoluminescent detectors.

The most significant finding was that the neutronic and economic performance of radial blanket assemblies are essentially independent of core size (rating) for radially-power-flattened cores. Hence the methodology and results of current experiments and calculations should be valid for the large commercial LMFBR's of the future.

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1.1 Foreword

This is the sixth annual report of the LMFBR Blanket Physics Project. This report covers work done since the last progress report, Reference (1), during the period from July 1, 1974 through June 30, 1975.

The MIT Blanket Research Project is part of the ERDA LMFBR development program, having as its primary objective the experimental investigation of clean, but realistic, benchmark mockups of the blanket-reflector region of large LMFBR reactors. The key experimental tool used in this work is the Blanket Test Facility at the MIT Research Reactor. The BTF contains a fission-converter plate tailored to deliver a neutron spectrum simulating LMFBR core leakage, which can be used to drive fast reactor blanket-reflector mockups.

Blanket subassemblies are constructed of uranium metal fuel rods, clad in carbon steel, surrounded by anhydrous sodium chromate. The homogenized mixture closely simulates UO_2 fuel, stainless steel clad and sodium metal coolant; all of the important heterogeneous effects are also closely simulated.

To date, four blankets have been investigated. Blanket No.1 was a borax-iron assembly used only for preliminary tests of system design performance; No.2 was a 3-subassembly-row, steel-reflected mockup of a typical large (1000 MWe) LMFBR design; and No. 3 was a 2-row, graphite-reflected mockup of an

advanced design. Blanket Mockup No. 4 was similar to No. 2, except that the converter was modified to drive it with a spectrum typical of a smaller, demonstration-reactor-sized core.

1.2 Active Research Areas

The MIT Research Reactor, which was shut down for renovation in May 1974, remained inoperable during 1975 because of a number of unforeseen difficulties. Hence it was not possible to resume blanket experiments according to our original schedule, and instead a modified program with a heavier reliance upon analysis, parametric and sensitivity studies was carried out.

During the report period work was carried out in the following areas:

1. The steel reflector assemblies required for Blanket Mockup No. 5 were designed and fabricated (see section 2.1 of the following Chapter). Experimental work was carried out on improved methods for gamma heating measurements and on analysis of data related to blanket breeding performance (also Chapter 2).
2. Analytic work was carried out on the potential use of graphite reflectors for LMFBR's, on internal blankets for LMFBR's, and on the economic aspects of blanket fuel management (Chapter 3).
3. A detailed evaluation of the use of state-of-the-art calculational methods for gamma heating in our blanket mockups was completed (Chapter 4).

4. The effect of increased reactor size on blanket neutronic performance was investigated (Chapter 5).

In the final chapter some general observations are made on the overall status of the project's efforts, and future research goals are outlined. Appendix A contains a complete listing of all project-related publications.

1.3 Staff

The project staff, including thesis students, during the report period was as follows:

M.J. Driscoll, Associate Professor of Nuclear Engineering
D.D. Lanning, Professor of Nuclear Engineering
V.A. Miethe, Computer Operations Assistant
A.T. Supple, Engineering Assistant
D.A. Bruyer, SM Student
C.A. Chambers, SM Student
O.K. Kadiroglu, ScD Student
M.S. Kalra, Research Assistant, ScD Student
M. Ketabi, SM Student
R.E. Masterson, Part-time employee, former SM Student
R.A. Morneau, Research Assistant, SM Student
J. Pasztor, Undergraduate Laboratory Assistant
R.A. Pinnock, Research Assistant, SM Student
A. Tagishi, Research Assistant (since Jan. 1974)
S.S. Wu, Nuclear Engineer's Degree Student

1.4 References

- (1) LMFBR Blanket Physics Project Progress Report No. 5
COO-2250-14, MITNE-170, June 30, 1974.

CHAPTER 2

EXPERIMENTAL WORK

2.1 Introduction

Despite the fact that the MIT Reactor was unavailable during the report period due to a prolonged shutdown for renovation, a number of experimental-related activities were carried out, including construction of a new steel reflector region which will facilitate neutronic and photonic studies in that region on Blanket Mockup No. 5. Work was also carried out on an improved TLD readout device based on the radiophotoluminescent response of ordinary LiF TLD's. And, finally data analysis on the backlog of information compiled on Blanket Mockup No. 4 continued. Each of these topics is addressed in the following subsections of this chapter.

2.2 Reflector Region Redesign

To facilitate reflector region measurements in subsequent blanket research we have constructed a new set of steel reflector assemblies to replace the laminated steel sheet used up until now (see Fig. 2.1 for an isometric view of this prior arrangement). Figure 2.2 illustrates a typical layout of the new reflector arrangement, and Figs. 2.3 and 2.4 show the design features of the various assemblies used to make up the new reflector region.

All assemblies are constructed of carbon steel, for which a representative composition is listed in Table 2.1

Fig. 2.1 Schematic View of Blanket Assembly with
Former Reflector Configuration

BLANKET NO. 5

Fig. 2.2 Blanket Mockup with New Reflector

- Key**
- 1 $2\frac{1}{2} \times 6 \times 48$
  - 2 $4 \times 6 \times 48$
  - 3 $3 \times 6 \times 48$
  - 4 $1\frac{1}{2} \times 6 \times 48$
  - 5 $1\frac{1}{2} \times 4 \times 48$
  - 6 $1\frac{1}{2} \times 3 \times 48$
  - 7 $\frac{1}{2} \times 2\frac{3}{4} \times 48$
  - 8 $\frac{1}{2} \times 1\frac{3}{4} \times 48$
  - 8 $\frac{1}{2} \times 1 \times 48$

$\frac{1}{2}'' \times \frac{1}{2}''$ Traversing Slots

Fig. 2.3 Horizontal Cross Section Through Reflector Assemblies

Fig. 2.4 Vertical Section of Typical Traversing Assembly (Sideview)

Table 2.1. New Reflector Composition

MATERIAL: Type C-1018 Steel, Cold Rolled

CHEMICAL COMPOSITION:

<u>CONSTITUENT</u>	<u>LOWER/UPPER, WT.%</u>
Carbon	0.15/0.20
Manganese	0.60/0.90
Phosphorous	0.04 max.
Sulphur	0.05 max.
Iron (remainder)	98.81/99.25

DENSITY: $490.58 \text{ lbs/ft}^3 = 7.86 \text{ gms/cc}$ IN SUBSEQUENT CALCULATIONS THE REFLECTOR WILL BE
TREATED AS PURE Fe WITH:

$$N = 0.08476 \text{ nuclei/barn cm}$$

Square cross section steel rods ($7/16"$ x $7/16"$) have been fabricated for insertion into the vertical square channels ($1/2"$ x $1/2"$) of the traversing assemblies. Circular spots have been milled into the surface of these rods to accomodate the various foils which will be irradiated in the reflector region. By rotating the holder 90° , the foils can be irradiated either perpendicular to or parallel to the "radial" axis of the blanket/reflector assembly. Vertical slots are also milled into the traversing rods to accomodate the standard TLD holders used for gamma heating measurements (1).

The initial set of runs to be carried out using these new assemblies will be U-238 (n,f) and In (n,n') foil traverses designed to resolve questions as to fast neutron penetration raised in prior work at MIT (2). This will be followed up by gamma heating measurements aimed at improving agreement between measured and calculated results developed by previous researchers at MIT (1,3).

2.3 TLD Reader Development

As noted in our previous experimental work (1), a continuing limitation on what can be achieved has been the precision attainable with a state-of-the-art TLD readout device.

Although the thermoluminescent dosimeter (TLD) is an established detector for both personnel dosimetry and gamma heating measurements, state-of-the-art methods have both practical and inherent deficiencies. Among the more troublesome problems are supralinearity and the destruction of the stored information during readout, neither of which are amenable to improvement through use of more advanced experimental techniques, such as photon counting.⁽⁴⁾

Our interest in improved methods for measuring gamma heating has led us to exploit the observation that LiF TLD's also exhibit radiophotoluminescent (RPL) response.⁽⁵⁾ A simple but effective readout apparatus has been developed which can be used for non-destructive readout of the RPL response using standard, commercially available TLD's.⁽⁶⁾

Figure 2.5 is a schematic diagram illustrating the following important features of the reader:

- (a) An intense source of white light: a GE Type 4537-2, 13 volt, 100 watt spotlight with built-in parabolic reflector. Passing the lamp beam through a pair of 4.5 inch diameter condenser lenses, provides a one cm. dia. spot of focused light to the filter assembly.
- (b) An inlet filter assembly consisting of, in succession:
  - 1) a 0.25 in. thick disc of heat-absorbing glass
  - 2) a thin aluminum mask having a 1 cm. dia. center hole

KEY

- ① LAMP
- ② CONDENSER LENS PAIR
- ③ INLET FILTER ASSEMBLY
- ④ LIGHT-TIGHT CAP
- ⑤ TLD HOLDER
- ⑥ TLD
- ⑦ COLLIMATOR
- ⑧ SLIDE-VALVE
- ⑨ OUTLET DIELECTRIC FILTER
- ⑩ PM TUBE

Side View: Not To Scale

Fig. 2.5 Schematic of RPL Readout Device for TLD's

- 3) a blue glass filter (1 mm thick Schott BG-12) transmitting a broad band of blue light centered at about 450 nm
  - 4) a dielectric interference filter centered at 450.0 nm $\pm 0.2\%$ (H.B.W. $\pm 0.15\%$, transmission outside passband $< 0.01\%$, peak transmission 50%)
  - 5) a second aluminum mask
  - 6) a short focal length lens to collimate the divergent transmitted light
- (c) The TLD holder: a spring-loaded split-collet chuck (adapted from a drawing pencil) which grips the upper 1.5 mm of the TLD and holds it rigidly vertical.
- (d) An outlet collimator: a 1/8 in. thick metal disc with a 1/8 in. dia. center hole.
- (e) An output dielectric interference filter centered at 650.0 nm $\pm 0.2\%$ (H.B.W. $\pm 0.15\%$, transmission outside passband $< 0.01\%$, peak transmission 50%).
- (f) The detector: an ITT FW-130-.1 type photomultiplier tube designed for single photon counting.
- (g) Associated Electronics:
- 1) A Fluke Model 402M High Voltage Power Supply to provide 1630 V to the PM tube
  - 2) A Mech-Tronics Model 511 Photon Discriminator to

amplify and pulse-shape the output signal

- 3) An EICO Model 1064S unregulated power supply to drive the lamp
- 4) A home-made voltage regulator circuit for the power supply. A simple series-pass unit using a Type 723 integrated circuit: capable of holding lamp voltage to its operating value of ~ 12 volts within ± 0.02 volts.
- 5) A Hewlett Packard 5381A Frequency Counter to display the output.

Adherence to good practice was found to be quite beneficial: performance was considerably enhanced by careful alignment and collimation of the optical path, by coating all interior surfaces with optical grade black paint, by rigid mounting of all components in the light path and by assiduous elimination of all light leakage. The light-tight enclosure containing the TLD was equipped with a cap to seal the loading port during readout, and a sliding gate valve to cut off the output signal during loading. A small fan was used to dissipate the heat deposited on the front face of the TLD enclosure and to cool the aluminum tube surrounding the PM detector. Care was taken in the selection and routing of cables to minimize pickup of electronic noise.

Harshaw LiF TLD-700 detectors (1mm dia., 6 mm long) were irradiated in a Co-60 calibration facility and used to evaluate the performance of the reader, with the following results:

- (1) excellent linearity was observed between response (counts/sec/mg) and exposure dose (rads) in the range examined ($0 - 10^4$ rads), consistent with the observation that RPL response does not exhibit the supra-linearity observed in TLD methods.
- (2) an acceptably low background was achieved for our purposes: unirradiated, annealed TLD's emitted a signal equivalent to approximately 100 rads exposure; under the same conditions the dark signal was equivalent to 10 rads. Background was the same for all TLD's within the reproducibility of the measurement technique.
- (3) reproducibility was also acceptable:
  - a) repetitive readings on the same TLD during the same loading confirmed the applicability of Poisson statistics, with σ^2 equal to the number of counts accumulated
  - b) repetitive reloads of a single TLD gave readings reproducible within approximately $\pm 2\%$
  - c) different TLD's irradiated to the same dose gave readings within $\pm 5\%$ of the mean value.

- (4) insignificant fading or drift was observed over a several day period (however normalization of all runs to a standard is commonly employed in observation of general good practice)
- (5) RPL readout was shown not to effect subsequent normal TLD readout of the same detectors
- (6) it is not clear whether normalization by weight improves precision: tests with irradiated TLD's having reduced length or diameter showed that the signal is less than directly proportional to weight, and in any event TLD's in a given batch typically weigh the same within $\pm 1.0\%$.
- (7) in our prototype reader a 1 mm dia., 6 mm long TLD exposed to 10,000 rad will give rise to approximately 10,000 counts in 10 seconds, a convenient value for our experiments, where many TLD's are typically involved in a given run.

The latest version of the readout device, described above, evolved via a sequence of analytic and trial-and-error improvements implemented by a series of investigators over an 18 month period.⁽⁶⁾ A variety of optical filters were tried on the inlet and outlet. Conventional glass filters were found to be unacceptable because of weak fluorescence by even selected varieties. A sequence of a dozen dielectric filters were tested on the output signal to vary the selected wavelength between 500 and 800 nm. The 650 nm filter gave the best signal-to-background

response, although signal magnitude was maximum near 520 nm, as expected from the prior work reported in Ref. (5); the inlet wavelength was also varied and 450 nm light shown to be optimum, in agreement with the literature^(5,7).

In general we conclude that RPL readout of TLD detectors is a useful experimental technique. We have been able to routinely achieve precision equal to or better than that of TLD readout. The major shortcoming is the moderately high background which makes measurement of doses less than about 10 rad impractical. It is likely, however, that further improvements could be effected. Use of a dye laser to excite the RPL response would greatly increase sensitivity (and cost) and would possibly permit a further reduction in background, as would finer tuning of the wavelength selected by the output filter. A limited amount of additional work is planned during the coming year to complete our evaluation of this device as a routine read-out method.

On a closely related matter, MIT participated in an intercomparison of TLD measurement methods with ANL and ORNL. The results have been reported by ORNL (8) and will not be repeated here. At the same time MIT intercalibrated its Co-60 standardization source and methods against ORNL's. Using MIT's TLD's irradiated both at MIT and ORNL in MIT's steel capsules gave a MIT source strength within 1% of the value established by MIT prior to the ORNL comparison. The comparison did uncover some aspects requiring further investigation: when MIT's TLD's were re-encapsulated at ORNL in one of their TLD holders, a 16% discrepancy in source strength

calibration was indicated, hence once a high precision readout device is routinely available we will investigate the effect of capsule design on TLD response; in addition, the ORNL experiments reported in (8) were at dose rates an order of magnitude below those normally experienced in MIT's prior experimental work, hence the need for more precise low-dose calibration has been identified if additional comparisons of this type are conducted.

2.4 Measurements of Breeding Performance

As noted in section 4.3 of Chapter 4 of last year's annual report (9), the most important single index of blanket breeding performance is the ratio $\bar{\sigma}_c^{28}/\bar{\sigma}_f^{25}$, where both numerator and denominator represent blanket-averaged values.

Because the MIT reactor was unavailable for experimental research during the period covered by this report, and will apparently not be available for some time to come, this work has been reprogrammed to include a substantial component of parametric and sensitivity studies. In view of this, we will postpone further discussion of this area until next year's annual report, which will then include substantial contributions on both analytical/numerical studies and new experimental data.

Preliminary results are showing that calculated values of $\bar{\sigma}_c^{28}/\bar{\sigma}_f^{25}$ are very insensitive to almost all variables except changes in the U-238 capture and U-235 fission cross sections themselves, and also that measured and calculated values are in relatively good agreement at ~ 0.11 .

2.5 References

- (1) P.A. Scheinert and M.J. Driscoll, Gamma Heating Measurements in Fast Breeder Reactor Blankets, COO-2250-10, MITNE-164, Aug. 1964.
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CHAPTER 3

BLANKET PERFORMANCE EVALUATION3.1 Introduction

During the past year work has continued in several areas which may be grouped under the general topic of blanket performance investigations. These include an assessment of the practicality of using graphite reflector assemblies for LMFBR blankets (1), additional work on the use of internal blankets for LMFBR's (2), and studies related to correlation of the breeding economics of fast reactor blankets (3,4). Work is also continuing on simpler ways to characterize the burnup history of fast reactor blankets (5). The subsequent sections of this chapter briefly summarize the results of these efforts.

3.2 Graphite-Reflected Blankets

Previous work at MIT (6) has shown potentially favorable aspects for the use of graphite reflectors for LMFBR blankets -- a circumstance which led to the work reported here, which focused more upon the engineering aspects of this concept.

After consideration of several alternatives, a reference design was developed for a graphite reflector assembly consisting of a stack of graphite discs in a cylindrical stainless steel can, inserted into a standard hexagonal assembly duct and cooled by sodium flowing up between the cylindrical can and the hexagonal duct. The graphite discs have their preferred

growth axis aligned vertically to permit accommodation of dimensional changes without overstressing the steel can. The axial segmentation also precludes the bowing problems which would occur if a single graphite log were used.

Thermal design analysis of the reference design showed that the maximum graphite temperature rise above ambient (1000 °F) was determined almost entirely by the gap between the graphite and the steel can, and could be limited to less than 900 °F by radiative heat transfer alone -- hence arbitrarily large radial shrinkage can be tolerated. In general the reference design appeared capable of meeting required thermal and mechanical design criteria.

Further analysis, however, identified a serious problem in that available HTGR-grade graphite, an obvious candidate for this application, would exceed neutron fluence levels covered by available test data in less than one year of exposure in the first row of reflector assemblies adjacent to a fast reactor blanket. Assured lifetimes on the order of 3 years may be obtainable using more expensive isotropic graphites, and periodic rotation of assemblies to uniformly distribute the damage could probably stretch the lifetime to about 5 years. If assembly life is less than ten or so years it will be difficult for graphite to compete with an added row of blanket assemblies or a conventional steel reflector assembly. Moreover, the modest neutronic advantages of using graphite probably do not justify

an extensive R&D program to qualify advanced graphites solely for this application. It should be noted, however, that graphite-steel laminae are proposed for use in the reflector region of the GCFR reference design; if successfully developed for this purpose, subsequent LMFBR applications could be contemplated.

In view of these findings we have suspended work on attempting to improve blanket performance through modifications in reflector design. It should also be pointed out that proposals to incorporate graphite moderator inside the blanket itself face an even greater problem, since fast neutron fluence can be an order of magnitude higher than in the reflector.

The best alternative to the use of high albedo reflector assemblies would appear to be installation of additional blanket assemblies, where the incremental deficit incurred is less than the cost of a reflector assembly: the use of ThO_2 fuel in these peripheral assemblies to produce premium value U-233 may increase the margin sufficiently to justify an entire extra row of blanket assemblies.

3.3 Internally-Blanketed Cores

Some additional work has been done (2) on the use of internal blankets in LMFBR's, following up on previous work at MIT by Ducat (7). The concept investigated is the so-called "parfait blanket", a disc-shaped region of depleted uranium fuel pellets at the midplane of the central zone of a two-enrichment-zone core.

Several variations on Ducat's basic design were investigated. The thickness of the external axial blankets was increased, thereby decreasing the thickness of the fissile-fueled region. This had the desired effect of reducing the fissile inventory of the parfait system but increased the peak power density, peak fast flux and reactivity swing over core burnup lifetime. Next the internal blanket was moved off-center toward the coolant outlet end of the core. This had the desired effect of reducing the peak clad temperature, but had undesirable side-effects: increased peak power density and slight increases in peak fast flux and plutonium inventory. Thus although it was shown feasible to engineer design trade-offs which mitigated certain disadvantages of the basic parfait design, the compensatory penalties suffered in other important criteria appeared to be excessive.

Simple models were developed to compare the susceptibility of parfait and conventional cores to thermal and fluence-induced bowing. It was found that the parfait design would experience maximum thermal and irradiation-induced bowing in the inner core which was only 42% and 16%, respectively, of that experienced in the conventional core. Because of this it was judged feasible to increase the volume fraction fuel in the inner core; in fact the axially-integrated radial power profile is sufficiently flat to permit considering the radical alternative of removing the hexagonal subassembly ducts in the inner zone. Increasing the central-zone fuel volume fraction (from its reference value of

30%) to 35% increased the mid-cycle breeding gain by 44% and decreased the burnup reactivity swing by 51%; a further increase to 40% fuel yielded a 67% increase in breeding gain and a 73% drop in reactivity swing. A breeding ratio of 1.40 can be obtained, compared to 1.20 for a conventional core.

We therefore conclude that the most productive trade-off on the advantages of the parfait design would be to increase the volume fraction fuel.

During the course of this work some parametric studies were also carried out of more general interest relative to blanket performance and its calculation.

Calculations were carried out on a representative reference 1000 MWe LMFBR (described in detail in Ref. (7)) using the 2DB program and a 4-group cross section set (again see (7)) to evaluate the extent of subdivision needed in burnup studies and simultaneously to extract information on radial blanket fuel management and design alternatives. All blankets investigated were driven by the same equilibrium core, which once-defined was not burned up further: the two-subassembly-row radial blankets were irradiated for 1800 days in 150 day time steps.

In the first series of runs it was shown that use of 1 instead of 6 axial zones (per half-core) in the radial blanket over-predicted Pu production by 5.1% and underpredicted blanket power production (at 1800 days) by 16.5%. This is attributed to the fact that zones are homogenized in 2 DB after each time step --

hence using only one zone in effect removes bred plutonium from the center to the axial extremities where the flux is much lower and reduces subsequent burnup. Nelson (8) did a similar study on the core and found that using 1 in place of 6 zones gave an 8% low reactivity loss: here the code moved fresh Pu from the extremities to the center -- the reverse of the process in the blanket.

The remainder of this work involved variation in the number of radial zones and the radial reflector composition. The results are shown in Table 3.1. Note that the homogenization by zone carried out by 2 DB can also be interpreted as an equivalent fuel management strategy: with 1 zone per assembly it corresponds to periodic rotation of the assembly; with 1 zone for the entire blanket it corresponds to periodic interchange of assemblies between rows (coupled with concurrent rotation).

Several interesting results are evident:

- (a) it is necessary to use several zones per assembly to accurately describe the radial blanket
- (b) assembly rotation by itself produces very little gain in fissile production
- (c) assembly interchange produces a small, but perhaps worthwhile increase in fissile production
- (d) use of a row of graphite reflector assemblies can effect a modest improvement in breeding performance: examination of the detailed data shows that 90% of the

TABLE 3.1 RESULTS OF RADIAL BLANKET BURNUP CALCULATIONS

<u>Case</u>	<u>Reflector</u>	<u>Number of Radial Zones</u>	<u>% Increased Pu*</u>	<u>% Increased Power*</u>	<u>Comment: equivalent</u>
1	Stainless Steel	4	--	--	Reference Case batch irradiation
2	Stainless Steel	2	0.41	-0.73	Assembly rotation
3	Stainless Steel	1	2.06	-4.55	Assembly interchan
4	Graphite (one row backed by SS)	4	4.45	7.37	Batch irradiation
5	Graphite (one row backed by SS)	2	4.76	7.00	Assembly rotation
6	Graphite (one row backed by SS)	1	5.97	4.46	Assembly interchan

All cases - 2 row radial blanket, depleted uranium loading, 6 axial zones (half core)

* 1800 days exposure

improvement comes from the assemblies adjacent to the reflector (see, however, Section 3.2 of this Chapter which deals with practical difficulties involved with the use of graphite in FBR's).

While these results are not entirely new, (e.g. see Ref. (9)), they set the stage for a continuing series of runs in which first blanket material composition and then cross sections will be modified.

3.4 Breeding Economics

Work has continued to refine analyses of the economic aspects of blanket performance. The work discussed here was completed subsequent to the analyses carried out by Tagishi (10), which are summarized in Chapter 5 of this report. The present analysis was carried out in two steps: first the theoretically interesting case of linear fissile buildup with time was studied (3), then the linear results were modified to accomodate cases of practical interest, which involve (slightly) non-linear fissile buildup histories.

The procedure followed involved use of series expansions and approximations in the exact cash flow model (as used in the BRECON code) to obtain analytic equations for the breakeven and optimum irradiation times for cases involving linear buildup of fissile inventory as a function of time. These results were then compared with exact results from BRECON. Next higher-order terms were retained in the analytic model to obtain corrections for non-linear buildup. Results were again compared with BRECON.

The cash flow expression for the fuel cycle cost contribution of a blanket region takes the form:

$$e \propto \frac{C_1 F_1(-T_1) + C_2 F_2(-T_2) - C_3 F_3(-T_3) E(T)}{T} \quad (3.1)$$

where

e is the fuel cycle cost contribution, mills/kwhr

C_i are cost components (\$/kg) for operation i
($i = 1, 2, 3$ refer to fabrication, reprocessing and material credit, respectively)

$F_i(-T_i)$ are carrying charge factors for operation i , occurring at time T_i measured from the irradiation midpoint = $\frac{1}{1-\tau} [(1+X)^{T_i} - \tau]$ with τ and X the tax and discount factors (fractions), respectively.

$E(T)$ is the fissile enrichment at time T , which is the duration of the irradiation.

Figure 3.1 shows the history of the cash flows associated with Eq. (3.1).

Series expansion of the F_i for linear $E(T)$ (i.e. $E(T) = ST$) leads to the following results for the case in which post-irradiation transactions are not capitalized (i.e. $\tau = 0$ for $i = 2, 3$):

- 1) Optimum irradiation time, $(de/dT) = 0$:

$$T_{OP} = (2W/SX)^{1/2}, \text{ yrs} \quad (3.2)$$

- 2) Breakeven irradiation time, $e = 0$

$$T_{BE} = W/S, \text{ yrs} \quad (3.3)$$

Fig. 3.1 Timing of Cash Flows for Fuel Cycle Cost Calculations

3) Minimum profitable fissile buildup rate

$$S_{\text{MIN}} = 2WX, \text{ kg fissile/kg HM per yr} \quad (3.4)$$

where

$$X = (1-\tau) f_b r_b + f_s r_s \quad (3.5)$$

and f , r are the bond(b) or stock(s) fractions (f) and rates of return (r), respectively

$$W = \frac{C_1 F_1 (-\Delta T_1) + C_2 F_2 (\Delta T_2)}{C_3 F_3 (\Delta T_3)} \quad (3.6)$$

S = region-averaged linear fissile buildup rate,
kg fissile/kg HM per year

Several interesting points are brought out by the above results. An eleven-parameter economic environment has been reduced to two: X and W . Further, from Eq. (3.3), W can also be identified as the enrichment at breakeven.

Because of the assumptions and approximations used in deriving Eq. (3.2) it gives absolute results which are roughly 10% less than the exact BRECON results. We can, however, use the form suggested by Eq. (3.2) to develop an empirical curve-fit to the exact results:

$$T_{\text{OP}} = 538 \left(\frac{W}{S} \right)^{0.527} X^{-0.445} \text{ days} \quad (3.7)$$

Equation (3.7) reproduces the optimum irradiation time within an average error of only five days for periods averaging six years and for a wide range of all eleven economic parameters.

Similarly, for the breakeven time: the relation

$$T_{BE} = 422 \left(\frac{W}{S} \right)^{1.054} X^{0.11} \text{ days} \quad (3.8)$$

is accurate within an average error of 16 days over periods averaging 1.5 years.

Reference (3) develops additional results of increasing empiricism and diminishing accuracy. The following are of some interest, however. Consider the case when post irradiation transactions are capitalized, then

$$T'_{OP} \cong T_{OP} \sqrt{1-\tau} \quad (3.9)$$

$$S'_{MIN} \cong 2 S_{MIN} \quad (3.10)$$

Equations (3.9) and (3.10) show that (for $\tau=0.5$, a representative value for private utilities) the optimum irradiation schedule is 30% shorter and the minimum profitable buildup rate is twice as large. This has a profound impact on the fuel management and design (i.e. selection of optimum thickness) of fast reactor blankets. Hence the currently unresolved accounting question of whether post-irradiation transactions should or should not be capitalized deserves closer attention.

Finally, it is relatively easy to show that fuel-managed blankets (in-out, out-in, batch, zone-scatter) obey the above relations if the position and time averaged value of S is employed.

The next step in extension of the preceding results to more realistic cases involved consideration of non-linear fissile buildup histories, $E(T)$. By assuming that $E(T)$ was composed of the product of a linear term and an exponential function of time, and retracing the various steps in the derivation, it was possible to develop corrections to the linearized model.

One obtains:

$$\left[\frac{T_{OP}}{T_{OP,L}} \right] = \left[1 + \frac{1}{\bar{X}T_{OP,L}} \left(\frac{\Delta m}{m} \right) T_{OP,L} \right]^{-1/2} \quad (3.11)$$

and

$$\left[\frac{T_{BE}}{T_{BE,L}} \right] = \left[1 - 2 \left(\frac{\Delta m}{m} \right) T_{BE,L} \right]^{-1} \quad (3.12)$$

where the $\frac{\Delta m}{m}$ term is the deficit (or surplus) fissile mass accumulated at time T relative to the linearized model. Figure 3.2 illustrates the relationships involved in evaluation of the various terms in Eqs. (3.11) and (3.12): one merely solves for the linear model times using Eqs. (3.7) or (3.8), determines $\Delta m/m$ from a burnup history, and enters the results into Eqs. (3.11) or (3.12).

The above equations were applied to fissile buildup histories calculated for representative LMFBR blankets, both UO_2 and UC fueled. While errors up to 50% could be incurred in the prediction of T_{OP} using the linear model, Eq. (3.11) was correct to within

Fig. 3.2 Calculation of $\Delta m/m$ for Use in the Correction Factor

approximately one month for T_{OP} values averaging about 6 years. The correction to $T_{BE,L}$ was generally negligible.

Work is continuing in this general area. The primary focus is upon the correlation of blanket burnup histories using only beginning-of-life physics calculations. This would allow one to carry out blanket fuel management calculations without repeating detailed burnup histories for each case considered. Although some prior work (5) has shown that higher order perturbation theory may be well suited for this task, at present it appears that even simpler flux-time correlation methods may suffice for this purpose.

3.5 References

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Chapter 4

GAMMA HEATING CALCULATIONS

The work summarized in this chapter will be reported in detail in the topical report:

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"Gamma Heating in LMFBR Media"

COO-2250-18, MITNE-179

(to be published late 1975/early 1976)

In this work we have attempted to evaluate the utility of state-of-the-art methods for coupled neutron-gamma calculations available to us in the early 1970's. While the major focus has been on their applicability to analysis of gamma heating measurements carried out on fast reactor blanket mockups at MIT, sufficient generality has been maintained to make the results of interest to others in the FBR field including those having different objectives in mind.

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" (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. (2).

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 (3).

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 (4) and the LASL 42-group (5) coupled P_3 neutron-gamma cross section libraries, and a one-dimensional discrete ordinate transport code, ANISN (6), 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 (7). 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 (8).

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 (9,10). 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 (7):

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

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

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 (11). 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. 7).

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 (4) and the LASL 42-group (5) coupled neutron-gamma cross section sets, normalized to the same core peak power density of 700 Kw/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

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

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 the 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, $E_{Y,C}^1$ (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 (7).

fore be calculated as follows:

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

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 (7).

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

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 (12). 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 (2). 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

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

^bNormalization factor = 1.06

^cNormalization factor = 1.00

^dNormalization factor = 1.01

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.

Table 4.7

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.

4.6 References

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Chapter 5
THE EFFECT OF CORE SIZE ON
BLANKET PERFORMANCE

The work summarized in this chapter has been reported in detail in the topical report

A. Tagishi and M.J. Driscoll

"The Effect of Reactor Size on the Breeding
Economics of LMFBR Blankets"

COO-2250-13, MITNE-168, February 1975

This research was motivated by our need to decide how much attention should be given to changes in core (converter) characteristics in planning future blanket experiments. However the results have broader applicability in terms of their implications regarding the adequacy of computational methods, economic analyses, and technology transfer as fast reactors increase in rating over the next several decades.

5.1 INTRODUCTION

Commercialization of the LMFBR will require a continuous escalation in unit size to keep pace with the scale-up of competing nuclear or fossil alternatives: power ratings as large as 3000 MW_e are projected by the year 2000. The purpose of the investigation reported here has been to examine the effects of the required core size increases on the neutronic and economic performance of the breeding blanket region surrounding these cores. The approach used throughout has been to employ simple reactor physics and economics models to predict trends and develop correlations, and then to employ more sophisticated state-of-the-art computer methods to confirm the results for realistic system designs.

Figure 5.1 depicts the major material subdivisions in a two-dimensional representation of a 1000-MW_e LMFBR used as a reference design for burnup calculations. The main features to note in this cylindrically symmetric layout are two approximately equal-volume core enrichment zones (for radial power-flattening), a 40-cm-thick axial blanket on the top and bottom of the core, and a three-row, 45-cm-thick radial blanket surrounded by a steel reflector. Tables 5.1 and 5.2 summarize the pertinent data for this reference configuration, which closely resembles other 1000-MW_e LMFBR blanket studies (1, 2). This basic design was varied to examine

Fig. 5.1 Schematic Elevation View of the Upper Right Quadrant of the Standard 1000 MWe Reactor System

Table 5.1 Characteristics of the Standard 1000 MWe Reactor

Parameter		Dimension	Value
Core	Height	cm	100
	Diameter	cm	250
	Volume	liters	4906
Axial Blanket Thickness		cm	40
Radial Blanket Thickness		cm	45
Core Averaged Power Density		kwth/liter	~ 500
Core Peak Power Density		kwth/liter	~ 730
Core Zone I O.D.		cm	90
Core Zone I Volume		liters	2540
Core Zone II O.D.		cm	125
Core Zone II Volume		liters	2366
Rated Thermal Power*		MWt	2500
Rated Electrical Power**		MWe	1000

* Core Plus Blankets

** Assumed System Efficiency = 40%

Table 5.2 Standard 1000 MWe Reactor Parameters

	Ht. (cm)	Rad. Thick. (cm)	Percent by Volume		
			Fuel ^c	Coolant ^d	Structure ^e
Core					
Zone 1	100	90	30(85% t.d.)	50	20
Zone 2	100	35	30(85% t.d.)	50	20
Axial Blanket ^a	80	125	30(85% t.d.)	50	20
Radial Blanket ^b					
Row 1	180	15	50(95% t.d.)	30	20
Row 2	180	15	50(95% t.d.)	30	20
Row 3	180	15	50(95% t.d.)	30	20

Table 5.2 Standard 1000 MWe Reactor Parameters
(continued)

	Ht. (cm)	Rad. Thick. (cm)	Percent by Volume		
			Fuel ^c	Coolant ^d	Structure ^e
Axial Reflector ^a					
For axial blanket	50	125	-	50	50
For radial blanket	50	45	-	30	70
Radial Reflector	140	50		10	90

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

^b Assumes hexagonal assemblies 15 cm across the flats.

^c Fuel 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

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

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

reactor ratings in the range from 250 MW_e to 3000 MW_e. Most of the variations involved changing the core radius (from 65 cm to 215 cm) at constant core height (100 cm) while achieving an acceptable degree of radial power-flattening by means of radial enrichment zoning. For completeness, some studies were also carried out in which height was varied at constant radius. In all cases the blanket thickness was held constant. Tables 5.3 and 5.4 show the power ratings, the corresponding core radii and the enrichment zone configurations examined.

In all of the above cases, the blanket thickness and the irradiation time were optimized according to economic criteria determined by evaluating the levelized fuel cycle cost contribution of each blanket region, taking into account the burnup/economic parameters (including fissile revenue and fabrication, reprocessing and carrying charges).

Batch blanket management was selected for this study due to its simplicity of implementation (i.e., blanket elements see only one position in the reactor, minimizing reactor down-time devoted to blanket refueling and/or repositioning) and due to the fact that approximately the same amount of plutonium is bred in an equivalent number of blanket elements regardless of management scheme (e.g., out-in or in-out management) over the same time interval (see 1, 2 or 3).

Power Rating (MWe)	Core Radius (cm)	Core Volume (liters)	Incremental Core Volume (liters)	H/D_o *1	Number of Radial Enrichment Zones
250	65	1328	1212	0.77	1, 2, 6
500	90	2540	1260	0.56	1, 2
750	110	3800	1100	0.45	1, 2
1000	125	4900	2440	0.40	1, 2, 6
1500	155	7340	2640	0.32	1, 3
2000	180	9980	2380	0.28	1, 4
2500	200	12360	1956	0.25	1, 5
3000	215	14316		0.23	1, 2, 6

*1 Core Height = 100 cm

Table 5.3 Key Characteristics of Reactor Sizes Examined
(Constant Core Height)

Table 5.4 Key Characteristics of Reactor Sizes Examined
(Constant Core Radius)

Core Height (cm)	Core Volume (liters)	Core ^{#1} Average Power Density (kwth/liter)	Peak ^{#2} Power Density (kwth/liter)	H/D _o ^{#3}	Number of Radial Enrichment Zones
100	4900	510.2	1118.0	0.40	1, 2
150	7363	327.7	791.0	0.60	1, 2
200	9817	254.7	615.1	0.80	1, 2

*1 Total Reactor Thermal Power = 2500 MWth
Power Rating = 1000MW_e
Assumed Thermal Efficiency = 40%

*2 One Zone Core

*3 Constant Core Diameter = 250 cm

5.2 COMPUTER AND ANALYTIC MODELS

5.2.1 Method of Burnup

The primary calculational tool used in comparing the neutronic performance of the cases studied was the two-dimensional diffusion theory burnup code 2DB (4). A 4-group, regionwise cross-section set was employed. This set was prepared by collapsing the 26-group ABBN Type FTR-200 cross-section set (2, 5, 6) over spectra appropriate to the various reactor regions using the one-dimensional transport theory code ANISN (7). Regionwise 4-group cross sections have been shown to give results which compare favorably to multigroup many-region calculations, especially in the inner regions of the blanket which contribute most of the blanket plutonium production (8).

Since long burnups (around six years) were performed in studying the blanket burnup behavior, an "equilibrium" core and axial blanket were defined which remained fixed in time. Constant core and axial blanket material concentrations (including fission product and control poison concentrations) at the time-averaged values expected in these regions were used, obviating the need to fuel-manage the core. It should be noted that although considerable care was taken to define a realistic core and axial blanket composition around which the radial blanket was irradiated, other investigators (2, 8, 9) have looked into the effect of different core-management methods on radial blanket economics and have concluded that there is an insignificant effect. Furthermore, since the same core treatment is used for all cases studied, any systematic bias (however small) should cancel out so long as relative comparisons are emphasized.

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5.2.2 Depletion-Economics Model

The depletion-economic analysis was performed utilizing the cash flow method (CFM) contained in the computer code BRECON, developed by Brewer (1) and modified by Wood (2) to permit direct use of 2DB burnup results as input. This method capitalizes, and consequently depreciates for tax purposes, the material purchases and fabrication charges; whereas reprocessing charges and material credit are treated as an expensed cost and taxable revenue, respectively.

The economic results generated by BRECON are the local levelized fuel component of the energy cost (mills/kwhr) which can be applied to an entire region (e.g., radial blanket) or subregion (e.g., radial blanket row). This facilitates the determination of the minimum fuel cycle cost contribution (i.e., optimum irradiation time) for a blanket row or for the entire blanket.

Table 5.5 lists the basic economic parameters used in this study of radial and axial blanket burnup fuel economics. The conditions are the same as those employed by Brewer (1) and Wood (2) and are within the range projected for the mature U.S. nuclear fuel cycle economy (3).

5.2.3 One-Group Model

Simplified models were used to examine both the neutronic and economic performance of fast reactor blankets. Reactor physics properties were investigated using one-group diffusion theory and a representative set of fast reactor cross sections; economic

Table 5.5 Reference Economic Environment

<u>Operation</u>	<u>Unit Fuel Processing Costs, \$/kg</u>	
	<u>Axial Blanket</u>	<u>Radial Blanket</u>
Fabrication	80	69
Reprocessing	50	50

<u>Isotope</u>	<u>Isotope Market Value, \$/kg</u>
U-238	0
Pu-239	10,000
Pu-240	0
Pu-241	10,000
Pu-242	0

<u>Financial Parameter</u>	<u>Value of Parameter</u> <u>Private Utility</u>
Income tax rate, τ	0.5
Capital structure	
Bond (debt) fraction, f_b	0.5
Stock (equity) fraction, f_s	0.5
Rates of return	
Bonds, r_b	0.07
Stocks, r_s	0.125
Discount rate, X^*	0.08

$$*X = (1-\tau)r_b f_b + r_s f_s$$

performance was approximated using a simplified version of the cash-flow model employed in the detailed computer studies.

One-group diffusion theory was employed to predict the effect of changes in core size on both the radial and axial blankets. In each case, two types of cores were examined – uniformly enriched and radially power-flattened by enrichment gradation; and two independent variations in size were considered – increasing radius at constant core height, and increasing height at constant core radius.

The analysis of single-enrichment-zone cores is an elementary exercise dealt with in depth by all reactor physics texts; radially power-flattened systems are seldom discussed, despite their practical importance, and therefore the approximations required to permit an analytic solution are worthy of mention, the most important being that the core diffusion coefficient, the linear extrapolation distances into the blankets, and the core-averaged fertile-to-fissile fission ratio are essentially independent of local enrichment. Two-dimensional multi-group calculations were carried out to confirm each of these key assumptions and other common hypotheses, such as the spatial separability of the axial and radial flux dependence.

The subject reactor physics model was employed to develop the following conclusions:

1. For radially power-flattened (RPF) cores, both central and peripheral enrichments remain essentially the same as the core radius (hence power rating) increases for constant core height.
2. The magnitude of the peripheral core flux and leakage current (n/cm^2 -sec) into the radial blanket remains the same as

core radius increases; because the peripheral enrichment is the same, one also expects the energy spectrum to be the same.

3. The core-averaged enrichment (hence critical mass) for single-enrichment and RPF cores having the same height and radius (hence volume) are very nearly the same, and as expected, average enrichment decreases gradually with increasing core size.

4. Axial leakage is essentially the same for the RPF and one-zone cores.

Based on these observations, a number of important conclusions were drawn in regard to the rate of production of new fissile material in three blanket volumes of interest: the entire blanket, a single row, and per blanket assembly. Tables 5.6 through 5.8 summarize these results. The most important single conclusion is that per blanket assembly the fissile material bred per unit time is independent of core radius for RPF cores, but decreases rapidly for single-zone cores. Hence, in the more realistic RPF cores of commercial interest, the blankets will tend to have similar optimum thicknesses and optimum irradiation schedules and equal dollar revenue on a per assembly basis. On the other hand, the effect of an assembly row or an entire blanket on the total fuel cycle cost in mills/kwhr decreases as core size increases.

Table 5.6 Summary of the Effect of Changing Core Radius at Constant Core Height on the Neutronic and Economic Performance of the Radial Blanket (Predictions of Simple One-Group Model)

Core		Radially-Power-Flattened Core	Uniformly-Loaded Core
Items			
NEUTRONICS	Whole Blanket (or Row) Fissile Buildup Rate, $R_o(\text{kgPu}/\text{yr})$	$\propto a$	Constant $\cdot (1 + \frac{\delta_r}{a})^{-2}$
	Specific Inventory Buildup Rate $S_o(\text{kgPu}/\text{kgHM yr})$	Constant	$\propto a^{-2} (1 + \frac{\delta_r}{a})^{-2}$
	Breeding Ratio b_{xr}	$\propto a^{-1}$	$\propto a^{-2} (1 + \frac{\delta_r}{a})^{-2}$

a = Core Radius

δ_r = Extrapolation Distance

Table 5.6 Summary of the Effect of Changing Core Radius at Constant Core Height on the Neutronic and Economic Performance of the Radial Blanket (Predictions of Simple One-Group Model)

(Continued)

Core Items		Radially-Power-Flattened Core	Uniformly-Loaded Core
ECONOMICS	Revenue per Blanket Row (\$)	$\propto a$	$\text{Constant} \cdot (1 + \frac{\delta_r}{a})^{-2}$
	Revenue per Blanket Assembly (\$)	Constant	$\propto a^{-1} (1 + \frac{\delta_r}{a})^{-2}$
	Contribution to Fuel Cycle Cost (mills/kwhr)	$\propto a^{-1}$	$\propto a^{-2} (1 + \frac{\delta_r}{a})^{-2}$
	Optimum Irradiation Time (Full Power Days)	Constant	Increases
	Optimum Thickness (cm)	Constant	Decreases

a = Core Radius

δ_r = Extrapolation Distance

Core Items		Large Radially-Power-Flattened or Uniformly Loaded Cores
Neutronics	Whole Blanket Fissile Buildup Rate $R_o(\text{kgP}_u/\text{yr})$	$\propto a^2$
	Specific Inventory Buildup Rate $S_o(\text{kgP}_u/\text{kg}^{\text{HM}} \text{ yr})$	Constant
	Breeding Ratio b_{xa}	Constant
Economics	Revenue per Blanket Zone (\$/Blanket Zone)	$\propto a^2$
	Revenue per Blanket Assembly (\$/Blanket Assembly)	Constant
	Contribution to Fuel Cycle Cost (mills/kwhr)	Constant

a = Core Radius

Table 5.7 Summary of the Effect of Changing Core Radius at Constant Core Height on the Neutronic and Economic Performance of the Axial Blanket (predictions of simple one-group model)

Core Items	Radially-Power-Flattened or Uniformly-Loaded Cores
Radial Breeding Ratio b_{xr}	Constant
Axial Breeding Ratio b_{xa}	$\propto \frac{1}{H_e^2 \epsilon_c}$

$$H_e = H + 2\delta_a$$

H = Core Height

δ_a = Linear Extrapolation Distance

Table 5.8 Summary of the Effect of Changing Core Height at Constant Core Radius on the Breeding Performance of the Radial and Axial Blankets (Predictions of Simple One-Group Model)

5.2.4 Simplified Economic Analysis

The cash-flow model for the fuel cycle cost contribution of a given region of LMFBR blanket is of the form (1, 2):

$$e(T) \propto \frac{C_1 F_1(-T_1) + C_2 F_2(-T_2) - C_3 F_3(-T_3) \epsilon(T)}{T} \quad (5.1)$$

where

C_1 = fabrication charge, \$/kg heavy metal

C_2 = reprocessing charge, \$/kg heavy metal

C_3 = fissile value, \$/kg fissile

and the present worth factors are:

$F_1(-T)$ on the fabrication charges

$F_2(-T_2)$ on the reprocessing charges

$F_3(-T_3)$ on the material credit

The various times appearing in this relation are shown in Fig. 5.2, and a typical $e(T)$ curve is displayed in Fig. 5.3. Breakeven conditions occur when $e = 0$, and optimum performance results when $de/dT = 0$, as shown in the figure.

The computer code BRECON (1, 2) employs Eq. 5.1 to calculate regionwise cost contributions given time-dependent heavy metal inventories from a burnup code. Simple analytic solutions can be obtained, however, by series expansion (and truncation) of the present worth factors and by assuming a fissile buildup which increases linearly with time and decreases exponentially with distance from the core/blanket interface – assumptions amply confirmed by multigroup calculations. By means of this approximate

Fig. 5.2 Timing of Cash Flows for Fuel Cycle Cost Calculations

Fig. 5.3 Typical Variation of Fuel Cycle Cost Contribution from a Fast Reactor Blanket

treatment, it is possible to derive analytic expressions for the breakeven and optimum irradiation times for local slices or entire regions of a blanket. Quite often the results are quantitatively correct, but their major interest here is as the starting point for the development of correlation methods to systematize the results obtained from the BRECON program.

Tables 5.9 and 5.10 summarize the analytic expressions developed in this simplified approach. Figure 5.4 illustrates the network of linearized relations developed. The most important result is the simple power-law dependence of optimum and breakeven times and enrichments on but two parameters, W and X , which completely characterize the economic environment.

5.3 BEGINNING-OF-LIFE BREEDING PHYSICS RESULTS

5.3.1 Physics-Related Parameters

As has been noted, the approach employed in the present investigation was to develop hypotheses using simplified analytic models, and then to use the results of more sophisticated state-of-the-art computations to test assumptions and conclusions. In the paragraphs which follow, the results of this verification process are summarized for all of the major points at issue.

Figure 5.5 compares the critical enrichments predicted by the one-group model (1GM) and the 4-group 2DB calculations to provide a general picture of the degree to which the 1GM results can be relied upon. The figure also shows the close correspondence between single- and multi-zone, core-averaged critical enrichments.

	Local	Total
Breakeven	$\tau = \frac{1}{\gamma}(\ln T - \ln T_B)$	$\tau = \frac{2}{\gamma}(\ln T - \ln T_B)$
Optimum	$\tau = \frac{2}{\gamma}(\ln T - \ln T_O)$	$\tau = \frac{4}{\gamma}(\ln T - \ln T_O)$

T_B = The Interfacial Breakeven Time   Point P in Table 5.10
 T_O = The Interfacial Optimum Time   Point Q in Table 5.10
 γ = Reciprocal Diffusion Length of the Blanket

Table 5.9 Summary of Relations Between Local Blanket Region Positions or Total Blanket Thickness and Irradiation Time

Table 5.10 Summary of the Six Key Intercepts Predicted by Simplified CFM Analysis

Point	Curves	Thickness or Distance (CM)	Irradiation Time (Full Power Days)	Cost (mills/kwhr)
P	L-B T-B (Left)	0 0	$\frac{W}{S_o(0)}$ 196	0
Q	L-O T-O	0 0	$\sqrt{\frac{2W}{S_o(0)X}}$ 1121	0
R	L-B T-B (Right)	0 0	$\frac{2}{X} - \frac{W}{S_o(0)}$ 7464	0
M	L-B L-O	$\frac{1}{Y} \ln \left[\frac{S_o(0)}{2WX} \right]$ 45	$\frac{1}{X}$ 3750	N.A.
N	T-B T-O	$\frac{2}{Y} \ln \left[\frac{S_o(0)}{2WX} \right]$ 90	$\frac{1}{X}$ 3750	N.A.
S	L-B T-O	$\frac{2}{3Y} \ln \left[\frac{S_o(0)}{WX} \right]$ 48	$3 \sqrt{\frac{4W}{S_o(0)X^2}}$ 2224	N.A.

Fig. 5.4 Schematic Diagram of Breakeven and Optimum Loci as a Function of Irradiation Time

Fig. 5.5 Core-averaged Critical Enrichment as a Function of Core-size

In the analytic 1GM, perfect radial power-flattening was achieved by continuous enrichment gradation: in the multigroup calculations, a finite number of enrichment zones were employed. Table 5.11 summarizes the cases investigated. The various zones were selected to have roughly equal volumes, and zone enrichment was adjusted to match peak power densities in all of the zones to within ± 5 percent.

Figure 5.6 confirms that the central and peripheral enrichments of multizone cores tend to be the same within fairly narrow bounds. The conclusion that equal enrichment implies similar neutron spectra is confirmed by Fig. 5.7 which compares the peripheral neutron spectra of the 2-zone, 250-MW_e and 6-zone, 3000-MW_e cores.

5.3.2 Effect of Changing Core Radius at Constant Core Height on Blanket Performance

5.3.2.1 Radial Blanket

Figure 5.8 shows the BOL-specific fissile inventory buildup rate, $\bar{S}_O$, (gPu/kgHM-yr) of each row of the radial blanket driven by radially power-flattened cores, which substantiates the important conclusion that $\bar{S}_O$ is the same for all core radii (ratings) – hence the absolute economic performance of the radial blanket assemblies is the same for all cores.

It should also be noted that the $\bar{S}_O$'s for rows 1 and 2 (but not 3) are larger than the minimum profitable fissile buildup rate, which is defined as the rate beyond which a non-negative profit can be produced: here 2 gPu/kgHM-yr. The same results can also be rearranged to show that the percentage of the total blanket fissile material bred in a given

Power Rating (MWe)	Core* Radius (CM)	2 Zone Cores		Multi-Zone Cores					
		Zone:1	2	Zone 1	2	3	4	5	6
250	65	15.2	23.0	N.A.					
500	90	13.3	19.1	N.A.					
750	110	12.7	17.7	N.A.					
1000	125	12.5	17.2	11.5	12.1	13.0	14.1	16.8	22.0
1500	155	N.A.		11.8	13.0	16.4	~~17.0 17.9 20.0~~		
2000	180	N.A.		11.7	12.0	13.2	17.0	~~17.9 20.0~~	
2500	200	N.A.		11.7	11.6	12.2	13.5	17.9	~~20.0~~
3000	215	12.0	14.6	12.0	11.8	12.0	12.6	14.8	20.0

* Constant Core Height=100CM, All Cases

Table 5.11 Zone-Averaged Enrichments for Critical Beginning-Of-Life Cores of Various Ratings (2DB Calculations)

Fig. 5.6 The Effect of Core-Radius on Central and Peripheral or Central-Zone and Outermost-Zone Enrichment

ANISN - 26 Group, S_8 Calculations:
Neutron Flux at Interface Between
Core and Radial Blanket at B.O.L.

Fig. 5.7 Comparison of Peripheral Neutron Energy Spectra

Fig. 5.8 The Effect of Core-Radius on the BOL Fissile Specific Inventory Buildup Rate in the Radial Blanket

row is the same regardless of core radius: approximately 60% for row 1, 30% for row 2 and 10% for row 3, which is compatible with the equivalent-slab-geometry exponential-flux approximation used for the radial blanket in deriving some of the simple model results.

Figure 5.9 shows a log-log plot of the radial blanket breeding ratio as a function of core radius, which indicates that the log-log slope of the curve is -1.0, and thus the radial blanket breeding ratio is inversely proportional to core radius for RPF cores, as predicted by the simple 1GM. This implies that the relative importance of the radial blanket in fissile breeding decreases as the core radius increases, as expected. The more rapid decrease in blanket breeding ratio for one-zone cores is also illustrated.

5.3.2.2. Axial Blanket

Figure 5.10 shows the product of the core-averaged critical enrichment, ϵ_c , and the BOL-fissile specific inventory buildup rate, $\bar{S}_0$, of the axial blanket driven by either radially power-flattened or uniformly loaded cores, which indicates that the product ($\bar{S}_0 \epsilon_c$) is essentially constant, as predicted by the 1GM when one allows for the decrease in core-averaged enrichment with reactor size. The figure also shows that $\bar{S}_0$ increases slightly with the core radius but is very nearly constant for the larger cores investigated.

Figure 5.11 shows the product of the core-averaged critical enrichment and the axial blanket breeding ratio as a function of core radius, which indicates that the product is constant and the axial

Fig. 5.9 The Effect of Core Radius on Radial Blanket Breeding Ratio

Fig. 5.10 The Effect of Core-Radius on The 30L Fissile Specific Inventory Buildup Rate in the Axial Blanket

Fig. 5.11 Product of Axial Blanket Breeding Ratio and Core-Averaged Critical Enrichment as a Function of Core-Radius

blanket breeding ratio increases slightly with the core radius but is again very nearly constant for the larger cores examined.

5.3.2.3 Combined System Breeding Performance

Figure 5.12 shows the internal (core) breeding ratio, the blanket breeding ratio (sum of axial and radial blankets) and the total breeding ratio (core plus blanket) as a function of core radius for either radially power-flattened or uniformly loaded cores, which illustrates several interesting points in agreement with the predictions of the 1GM:

- 1) The internal (core) breeding ratio increases but does not exceed unity, even in the largest cores; the blanket breeding ratio decreases as core radius increases; the total breeding ratio remains very nearly the same. The total breeding ratio appears to be anomalously high (~ 1.5) because the results are for a BOL clean core without control poison or excess enrichment to compensate for burnup.
- 2) The internal breeding ratio for radially power-flattened cores is slightly less than for uniformly loaded cores because of slightly higher critical mass in the former.
- 3) The blanket breeding ratio for radially power-flattened cores is higher than that for uniformly loaded cores because of the higher radial breeding ratio due to radial power-flattening.

Fig. 5.12 The Effect of Core Size (Radius) on Breeding Ratio at Constant Core Height

5.3.3 Effect of Changing Core Height at Constant Core Radius on Breeding Performance

Although most reactor designers evidently prefer to scale-up LMFBR designs by increasing core radius at constant core height, for completeness, and as a further test of the 1GM predictions, a series of core height variations at constant radius were studied starting with the reference design 1000-MW_e, 2-zone core.

Figure 5.13 shows the radial blanket breeding ratio driven by either radially power-flattened or uniformly loaded cores as a function of core height, which indicates that the breeding ratio is the same regardless of core height. The predictions of the one-group model are in good agreement with the 2DB results.

Figure 5.14 shows the axial blanket breeding ratio driven by either radially power-flattened or uniformly loaded cores as a function of core height. The 2DB results confirm the 1GM prediction of a variation proportional to $(H + 2\delta_a)^{-2} \epsilon_c^{-1}$.

Figure 5.15 shows the internal breeding ratio, the blanket breeding ratio and the total breeding ratio as a function of core height, which indicates that:

- 1) The internal breeding ratio increases as core height increases, and ultimately exceeds unity; the blanket breeding ratio decreases as core height increases and the total breeding ratio remains very nearly constant.
- 2) The internal breeding ratio for radially power-flattened cores is less than that for uniformly loaded cores because of the higher critical mass of the former.

Fig. 5.13 The Effect of Changing Core Height at Constant Core Radius on the Radial Blanket Breeding Ratio

Fig. 5.14 The Effect of Changing Core Height at Constant Core Radius on Axial Blanket Breeding Performance

Fig. 5.15 The Effect of Changing Core Height at Constant Core Radius on the Breeding Ratio

- 3) The blanket breeding ratio for radially power-flattened cores is higher than that for uniformly loaded cores, but the total breeding ratio is the same regardless of the type of core loading.

Thus an increase in core height at constant core radius does not lead to an improvement in overall breeding performance.

5.4 DEPLETION-ECONOMICS RESULTS

Good agreement between the predictions of simplified analyses and 2DB-BRECON calculations has been obtained for the depletion-economics performance of LMFBR blankets. Useful correlations of blanket-breeding-performance characteristics, such as breakeven and optimum irradiation times and thicknesses, have been formulated using 2DB-BRECON calculations correlated against the parameters singled out in the simplified analysis.

Figure 5.16 shows the breakeven irradiation times for rows 1 and 2 and the optimum irradiation time for row 1 of the radial blanket as a function of core radius, which indicates that both breakeven and optimum times are the same for all core radii (power ratings) within half a refueling interval (assumed to be $\frac{1}{2} \times 300 = 150$ full power days). The agreement with the predictions of the simple model is therefore good on this important point.

Figure 5.17 shows the revenue per unit mass of blanket; the local (per blanket-row basis) and the total (entire-radial blanket) fuel cycle cost contribution as a function of the distance into the blanket

Fig. 5.16 The Effect of Core-Radius on the Optimum Irradiation Time for Row 1, and the Breakeven Irradiation Time for Rows 1 and 2, of the Radial Blanket

Fig. 5.17 Revenue per Blanket Assembly and Fuel Cycle Cost Contribution per Row or per Entire Batch-Irradiated Radial Blanket as a Function of Blanket Thickness (for a Given Irradiation Time)

(or the blanket thickness) for a given irradiation time, which indicates that the optimum thickness is the same for all core radii (power ratings) within half of a blanket assembly thickness (assumed to be $\frac{1}{2} \times 15 = 7.5$ cm).

Figure 5.18 shows the minimum fuel cycle cost contribution for each row as a function of core radius, which indicates that the minimum cost for a given irradiation time decreases approximately according to a^{-1} , i.e., inversely with core radius.

Table 5.12 summarizes the correlation coefficients for region j:

$$T = a_j W_j^{b_j} X_j^{c_j}, \quad (5.2)$$

comparing the predictions of the simplified model and the original results from the state-of-the-art computations using 2DB-BRECON. The range of parameters characterizing the economic environment used in obtaining the correlation is shown in Table 5.13. The utility of the simplified model in singling out the important parameters (W and X) for use in the correlations is evident. Figure 5.19 shows a typical comparison of correlated and original 2DB-BRECON results for the optimum and breakeven irradiation times for row 1 in the radial blanket driven by a radially power-flattened core (1000-MW_e, 2-zone). The correlation can predict the optimum and breakeven irradiation times within half of the refueling interval (here $\frac{1}{2} \times 300 = 150$ full power days).

Figure 5.20 compares for the four key correlation curves, predictions of the simplified model and the 2DB-BRECON results. The semilog linear relation between thickness and time is shown to be an excellent approximation, except for the "local breakeven" curve, which curves downward for large thicknesses and long irradiation time.

Fig. 5.18. The Effect of Core-Radius on the Minimum Fuel Cycle Cost Contribution From the Radial Blanket

			Simplified Model			2DB BRECON		
			a_j	b_j	c_j	a_j	b_j	c_j
BREAKEVEN	LOCAL	Row 1	2.64	$\times 10^4$		8.75	1.14	0.17
		Row 2	6.38	1.0	0.0	85.5	1.28	0.46
		Row 3	10.8					
	TOTAL	1 Row Blkt	2.64			8.75	1.14	0.17
		2 Row Blkt	38.2			34.6	1.30	0.28
		3 Row Blkt	53.4			75.3	1.34	0.40
OPTIMUM	LOCAL	Row 1	3.98	$\times 10^3$		9.22	0.54	-0.15
		Row 2	6.19	0.5	-0.5	7.75	0.40	-0.24
		Row 3	10.0					
	TOTAL	1 Row Blkt	3.98			9.22	0.54	-0.15
		2 Row Blkt	4.79			10.3	0.52	-0.18
		3 Row Blkt	5.66			10.2	0.50	-0.24

Table 5.12 Comparison of Correlation Coefficients Predicted by the Simplified Model to Those Obtained by Curve-Fitting 2 DB-BRECON Results

Parameter (Units)	This Study	Wood (W2)
Value of Fissile P_u (\$/g P_u)	6.0 to 14.0	6.0 to 16.0
Fabrication Charges (\$/kgHM)	69 to 140	69 to 140
Reprocessing Charges (\$/kgHM)	50 to 100	50 to 100
Discount Rate (Year ⁻¹)	0.04 to 0.16	0.075 to 0.085

Table 5.13 Range of Variation of Economic Parameters

Fig. 5.19 Comparison of Optimum and Breakeven Irradiation Times for Radial Blanket - Row 1 Predicted by Correlation and 2DB-BRECON

Fig. 5.20 Burnup Economics Correlation Lines Showing Key Loci

Table 5.14 compares the global optimum-point correlation coefficients predicted by the simplified model and those extracted by curve-fitting the data, indicating that the model can again be used to formulate useful correlations.

Figure 5.21 compares correlated and original 2DB-BRECON results for the global optimum thickness and the optimum irradiation time for various economic environments. The correlation can predict the global optimum thickness within a half-thickness (here 7.5 cm) of a blanket assembly, and the global optimum irradiation time within half of a refueling interval (here $\frac{1}{2} \times 300 = 150$ full power days).

Figure 5.22 compares correlated and original 2DB-BRECON results for the global minimum fuel cycle cost contribution. The agreement is good except for one point corresponding to an extreme economic environment (high Pu price and low discount rate).

Finally, it is concluded from the observations in this and the previous chapters that these correlations are applicable to an entire class of LMFBR core/blanket combinations characterized by constant core height and radial-core-power-flattening.

Table 5.14 Comparison of Correlation Coefficients for the Global Optimum:
Simplified Model vs. 2DB-BRECON

	Simplified Model			2DB-BRECON		
	a_j	b_j	c_j	a_j	b_j	c_j
T_s (Full Power Days)	1.77×10^3	+0.33	-0.67	1.53×10^3	+0.23	-0.45
τ_s (CM)	1.15×10^{-1}	-0.67	-0.67	1.38×10^{-2}	-0.86	-0.82
e_s (mills/kwhr)	N.A.			-1.50×10^{-7}	-2.46	-0.94

Fig. 5.21 Comparison of Global Optima Predicted by Correlation and 2DB-BRECON

Fig. 5.22 Comparison of Global Optimum Costs Predicted by Correlation and 2DB-3RECON

5.5 CONCLUSIONS

The purpose of this work – to examine the effect on blanket breeding economics of projected increases in core size in future commercial fast reactors – has been achieved using both simple models to elucidate general principles and state-of-the-art computer methods to provide realistic examples.

A one-group diffusion theory analysis of the radially power-flattened core behavior was employed to show that:

- (a) The breeding ratio of the radial blanket is, to a very good approximation, inversely proportional to the core radius for constant-height cores.
- (b) The fissile buildup rate per unit volume (hence per unit fertile mass or per blanket assembly) in the radial blanket remains constant as core size increases.
- (c) The total fissile production rate of the entire radial blanket increases as core size increases.
- (d) The peripheral core enrichment remains very nearly constant as the core radius increases. Hence the energy spectrum of neutrons leaking into the blanket will also not change significantly.

Translating these neutronic findings into their economic consequences, one finds that:

- (a) Total dollar revenue per radial blanket assembly in a given row of the blanket remains constant as core size increases.
- (b) Optimum blanket thickness remains constant as core size increases.

- (c) Optimum blanket assembly irradiation time remains the same as core size increases.
- (d) Total blanket dollar revenue increases as core size increases.
- (e) The fuel cycle cost contribution of the radial blanket, in mills/kwhr, decreases as core size increases. Hence the blanket does become a relatively less important economic entity.

Correlations for the breakeven and optimum irradiation times for a given row or for an entire blanket under batch fuel management, and the optimum blanket thickness have been developed which can predict time within half of a refueling interval and thickness within half of an assembly thickness. These correlations involve variables, and take the form, suggested by application of a simplified economics model:

$$T = a_j W^{b_j} X^{c_j} \quad (5.3)$$

$$\tau = \frac{1}{\gamma} \ln [a_j W^{b_j} X^{c_j}] \quad (5.4)$$

where

a_j , b_j and c_j are the correlation coefficients of the j -th region of the blanket,

γ is the inverse diffusion length of the blanket,

W is an economic parameter, and

X is the discount rate.

These conclusions were all tested against state-of-the-art burnup-economics calculations carried out on cores in the range

250 to 3000 MW_e using the 2DB depletion program and a 4-group cross-section set, and the BRECON cash-flow-economics program. Realistic multizone cores were examined: 2 radial enrichment zones up to 1000 MW_e and progressively more for the larger ratings. The simple model predictions were all confirmed to within practically significant accuracy: for example, as the core radius was increased, optimum blanket assembly irradiation times remained constant to within plus or minus half of a core refueling interval.

Similar analyses were carried out for axial blanket performance and for the effect on both the axial and radial blankets of changing core height at constant radius, and generally similar conclusions were reached. We also showed that for the single-enrichment-zone cores the optimum radial blanket thickness would decrease with core size.

These results are significant because they indicate that on a per assembly basis, radial blanket performance on demonstration-size units will be an excellent simulation of blanket performance on future commercial units, even in very large reactors. Likewise, fuel management schemes developed now for the smaller units will be directly applicable to the larger units; designers will not have to continuously re-optimize blanket design as fast breeder reactors grow in size to take advantage of the economics of scale.

5.6 RECOMMENDATIONS

In fulfilling the goal of the present work, several areas have been identified in which further analysis is merited, namely:

(1) The simple correlations developed here for the radial blanket, relating breakeven and optimum irradiation times and blanket thicknesses to the economic parameters W and X , should be extended to:

- (a) Facilitate their extension to different core and blanket designs (e.g., carbide fuel, different fuel fractions, axial blankets) with the minimum of new calculations at the 2DB-BRECON level. For example, a BOL snapshot calculation of the breeding ratio may suffice to properly normalize the correlations. It would also be extremely interesting to determine whether the exponents (b, c) in the time-thickness-economics correlations involving aW^bX^c are universal in the sense that they remain invariant for different core and blanket types, with the entire difference being accommodated in the "a" coefficient.
- (b) Develop a better correlation for the fuel cycle cost contribution e , mills/kwhr. This would eliminate the need for a detailed economics program (e.g., BRECON). This should be feasible since the purely empirical version examined in this report is already quite successful.
- (c) Deal with core economic performance. If this can be done, then it should also be possible to generalize the treatment to handle LWR or HTGR cores as well.

(2) A comparative analysis of different blanket fuel management options (e.g., zone-scatter, in-out, out-in) should be carried out using the methods of this report. The demonstration that blanket assembly performance will be independent of core size creates considerable

incentive to devote more attention to the fine points of blanket management now, since the results will not become outdated.

(3) The results of the present analysis have important implications for experimental programs such as those carried out on blanket mockups at MIT. For example, it does not appear very useful to study blankets driven by a variety of different (simulated) cores: as we have seen, the spectrum driving the radial blanket does not change significantly with core size, and that driving the axial blanket is bracketed by the essentially fixed central and peripheral spectra. On the other hand, this work motivates even more attention to in-depth analysis of a small number of representative blankets, which can serve as models for essentially all future applications.

(4) The methods of this report should be applied to cores in which some of the radial power-flattening is achieved by radial control poison gradation, as this approach is likely to be used in addition to, or even in lieu of, extensive radial enrichment zoning.

(5) This report has concentrated on the neutronic and economic aspects of blanket breeding performance. Some further consideration of other aspects of blanket design should be reviewed in light of its conclusions. For example, the finding that the magnitude and energy spectrum of the neutron flux in the radial blanket remains invariant as the core radius is increased also implies that assembly bowing due to stainless steel swelling will remain the same in the blanket (since swelling is roughly proportional to flux squared) - in contrast to the core

where the average flux increases as core size increases (due to the concurrent decrease in average enrichment). In another area, Brown (B3) has shown that an economic penalty should be assigned to blanket overcooling, to account for the resulting decrease in the core-averaged mixed-mean coolant temperature. As core radius increases, the ratio of the number of blanket to core assemblies decreases; hence the relative impact of blanket overcooling is also less.

(6) Throughout this analysis, we have employed Brewer's accounting method "A" in which material purchases and fabrication charges were capitalized and consequently depreciated for tax purposes; whereas reprocessing charges and material credit were treated as an expensed cost and taxable revenue, respectively. More work needs to be done in this area of dealing with appreciating assets such as the blanket and, if method A cannot be agreed on as a definitive convention, Brewer's method "B" should also be applied, and the results correlated using the methods of this report to determine whether blanket design and management options such as optimum thickness and irradiation time are significantly affected when compared to method A.

Finally, while an appreciable menu of future tasks has been suggested, it should not be considered as detracting from the conclusive nature of the demonstration presented here that large commercial fast reactors will have blankets which are in all essential respects the same as those found to be optimum for smaller demonstration reactors.

5.7 References

- (1) Brewer, S. T., E. A. Mason, and M. J. Driscoll, "The Economics of Fuel Depletion in Fast Breeder Reactor Blankets," COO-3060-4, MITNE-123 (November 1972).
- (2) Wood, P. J. and M. J. Driscoll, "Assessment of Thorium Blankets for Fast Breeder Reactors," MITNE-148, COO-2250-2 (July 1973).
- (3) Westinghouse Advanced Reactor Development, "Design Analysis of the LMFBR Radial Blanket," WARD-3045-T2B-9 (March 1973).
- (4) Little, W. W., Jr., and R. W. Hardie, "2DB User's Manual, Revision 1," BNWL-831, Rev. 1 (August 1969).
- (5) Abagyan, L. P., et al., Group Constants for Nuclear Reactor Calculations, I.I. Bondarenko (Ed.), Consultants Bureau, New York (1973).
- (6) Nelson, J. V., "Cross Sections for Preliminary Design of FTR, FTR Set No. 200," HEDL-TME-71-65 (1971).
- (7) Engle, W. W., Jr., "A Users Manual for ANLSDN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1693 (March 30, 1967).
- (8) Hoover, J., and D. A. Menley, "Alternative Neutron Energy Group Collapsing Schemes Applied to Fuel Cycle Calculations," ANL-7710 (January 1971).
- (9) Barthold, W. P., "Fuel Shuffling in LMFBR Blankets," Applied Physics Division, Argonne National Laboratory, FRA-TM-40 (August 1972).

CHAPTER 6

SUMMARY, CONCLUSIONS AND FUTURE WORK6.1 Introduction

This is the sixth annual report of the LMFBR Blanket Physics Project at MIT. During the past year work has been concerned primarily with the following:

- 1) Calculations of gamma heating rates in Blanket Mockup No. 4 for comparison with prior experimental results, and development of improved methods for TLD readout.
- 2) Investigation of the potential effect of reactor rating (core size) on blanket performance, and improved analyses of the economic aspects of blanket fuel management.
- 3) Additional analyses of blanket concepts which may have certain performance advantages, including internal blankets and graphite-reflected blankets.
- 4) Work in preparation for the resumption of experimental operations, including fabrication of an improved reflector region configuration, and analysis of data accumulated on Mockup No. 4 prior to shutdown.

6.2 Discussion

The most important conclusions which may be drawn from the past year's work are as follows:

- 1) Radial blankets driven by radially-power-flattened cores have essentially invariant performance characteristics

regardless of core radius (power rating). The magnitude and energy spectrum of the net current into the blanket remain the same, hence fissile buildup rates and breakeven and optimum irradiation times are unchanged. Since the central and peripheral enrichment of the core remain roughly the same as radius is increased at constant height, the volume-average performance of the axial blanket is also insensitive to core rating. In view of these findings considerably less will have to be done on experiments, methods development and reactor-specific studies than might at first appear the case.

- 2) The scope for improving blanket performance appears to be narrower than initially hoped for. Graphite reflectors, which were previously identified as having potentially favorable effects on blanket neutronics, have been shown to have difficulty sustaining the intense fast neutron damage to which they are unavoidably subjected. On the other hand, additional favorable evidence was developed of the beneficial effect of using internal blankets to improve overall core performance.
- 3) In the area of gamma heating it was shown that more emphasis needs to be placed in preparation of coupled neutron-gamma cross section sets on assurance of an energy balance. Conversely, less concern need be

placed on matching the spectrum of gamma sources or on use of a large number of gamma energy groups. Except near the interfaces between regions of highly dissimilar composition it was found that local production of gamma energy could be equated to local deposition. On the experimental front considerable progress was made on devising a radiophotoluminescent readout device for TLD's. Among the many advantages of this approach is the fact that the readout is non-destructive and hence may be followed by normal TLD readout (which is destructive).

- 4) Considerable progress was made on the development of correlation parameters for the economic analysis of LMFBR blanket management strategies. It was shown that an eleven-parameter characterization of the economic environment could be reduced to but two lumped parameters, which could then be used as the basis for correlation of breakeven and optimum irradiation times. Correlations of this type were studied for both linear and simple non-linear fissile buildup histories. Sufficient insight has now been gained to permit development of a combined correlation of economic and neutronic performance which will allow simple and rapid determination of blanket fuel management decisions.

6.3 Future Work

During the coming contract year, July 1, 1975 through September 30, 1976 (including transition period in U.S. Government FY) work will have to be arranged to fit the projected schedule for the resumption of MITR operations.

Priority items for future experimental work will include:

- 1) Completion of high-precision measurements on Blanket Mockup No. 4 necessary to assessment of our state-of-the-art capability to predict breeding performance.
- 2) Initiation of experiments on Blanket Mockup No. 5, which will have steel reflector assemblies (in place of the laminated steel sheet in Mockup No. 4), and can thereby facilitate investigation of the discrepancies in neutron and gamma transport noticed in our work on previous blanket mockups.
- 3) Completion of our work to develop an improved TLD readout device based on the RPL effect.

In the interim, increased emphasis will be placed on analysis, parametric and sensitivity studies, particularly those concerned with breeding performance, which has become the focus of considerable current interest.

Appendix A

BIBLIOGRAPHY OF BLANKET PHYSICSPROJECT PUBLICATIONS

In this appendix are tabulated all publications associated with work performed in the MIT Blanket Physics Project. Sc.D. and Nuclear Engineer's theses are listed first, followed by S.M. and B.S. theses and then by other publications.

A.1 Doctoral and Engineer's Theses

(Also see section 3 for corresponding topical reports.)

Forbes, I.A.

Design, Construction and Evaluation of a Facility for the Simulation of Fast Reactor Blankets, Feb. 1970

Sheaffer, M.K.

A One-Group Method for Fast Reactor Calculations, Aug. 1970.

Tzanos, O.P.

Optimization of Material Distributions in Fast Breeder Reactors, Aug. 1971.

Kang, C.S.

Use of Gamma Spectroscopy for Neutronic Analysis of LMFBR Blankets, Nov. 1971.

Leung, T.C.

Neutronics of an LMFBR Blanket Mockup, Jan. 1972

Ortiz, N.R.

Instrumental Methods for Neutron Spectroscopy in the MIT Blanket Test Facility, May 1972.

A.1 Doctoral and Engineer's Theses (continued)

Brewer, S.T.

The Economics of Fuel Depletion in Fast Breeder
Reactor Blankets, Oct. 1972

Gregory, M.V.

Heterogeneous Effects in Fast Breeder Reactors, Dec. 1972.

Wood, P.J.

Assessment of Thorium Blankets for Fast Breeder Reactors,
July 1973.

Ducat, G.A.

The Parfait Blanket Concept for Fast Breeder Reactors,
Jan. 1974.

Brown, G.J.

Evaluation of High-Performance LMFBR Blanket
Configurations, May 1974.

Scheinert, P.A.

Gamma Heating Measurements in Fast Breeder Reactor
Blankets, (Engineer's Thesis), Aug. 1974.

Tagishi, A.

The Effect of Reactor Size on the Breeding Economics
of LMFBR Blankets, Feb. 1975

Kalra, M. S.

Gamma Heating in Fast Reactors, Jan. 1976 (est.)

A.2 S.M. and B.S. Theses

Ho, S.L.

Measurement of Fast and Epithermal Neutron Spectra
Using Foil Activation Techniques

S.M. Thesis, MIT Nucl. Eng. Dept., Jan. 1970

Mertens, P.G.

An Evaluation of a Subcritical Null-Reactivity Method
for Fast Reactor Applications

S.M. Thesis, MIT Nucl. Eng. Dept., May 1970

Westlake, W.J.

Heterogeneous Effects in LMFBR Blanket Fuel Elements

S.M. Thesis, MIT Nucl. Eng. Dept., June 1970

Shupe, D.A.

The Feasibility of Inferring the Incident Neutron
Spectrum from Prompt Capture Gamma-Ray Spectra

S.M. Thesis, MIT Physics Dept., Aug. 1970

Pant, A.

Feasibility Study of a Converter Assembly for Fusion
Blanket Experiments

S.M. Thesis, MIT Nucl. Eng. Dept., Jan. 1971

Passman, N.A.

An Improved Foil Activation Method for Determination
of Fast Neutron Spectra

S.M. Thesis, MIT Nucl. Eng. Dept., Jan. 1971

Forsberg, C.W.

Determination of Neutron Spectra by Prompt Gamma-Ray
Spectrometry

M.S. Thesis, MIT Nucl. Eng. Dept., June 1971

Brown, G.J.

A Study of High-Albedo Reflectors for LMFBRs

S.M. Thesis, MIT Nucl. Eng. Dept., March 1972

Thompson, A.M.

Activation Profiles in Reactor Fuel Elements

B.S. Thesis, MIT Physics Dept., June 1972

Lal, D.

Determination of the Neutron Spectrum in the MITR
Transistor Irradiation Facility

B.S. Thesis, MIT Chem. Eng. Dept., June 1972

Ho, S.Y-N.

Selection of Foil Materials for LMFBR Neutron
Spectrometry

S.M. Thesis, MIT Nucl. Eng. Dept., May 1973

Choong, T.P.

Fast Neutron Spectrometry in an LMFBR Blanket Reflector

S.M. Thesis, MIT Nucl. Eng. Dept., Aug. 1973

Kennerley, R.J.

Proton-Recoil Neutron Spectrometry in a Fast Reactor
Blanket

S.M. Thesis, MIT Nucl. Eng. Dept., Aug. 1973

Chan, J.K.

A Foil Method for Neutron Spectrometry in Fast Reactors

S.M. Thesis, MIT Nucl. Eng. Dept., Jan. 1974

Yeung, M.K.

A Stacked-Foil Method for Epithermal Neutron Spectrometry

S.M. Thesis, MIT Nucl. Eng. Dept., Dec. 1974

Masterson, R.E.

The Application of Perturbation Theory and Variational
Principles to Fast Reactor Fuel Management

S.M. Thesis, MIT Nucl. Eng. Dept., Sept. 1974

A.2 Continued

Pinnock, R. A.

Parfait Blanket Configurations for Fast Breeder Reactors, S.M. Thesis, MIT Nucl. Eng. Dept., June 1975.

Chambers, C. A.

Design of a Graphite Reflector Assembly for an LMFBR S.M. Thesis, MIT Nucl. Eng. Dept., Aug. 1975.

Ketabi, M.

The Breeding-Economic Performance of Fast Reactor Blankets, S.M. Thesis, MIT Nucl. Eng. Dept., May 1975.

Bruyer, D. A.

Breeding-Economics of FBR Blankets Having Non-Linear Fissile Buildup Histories, S.M. Thesis, MIT Nucl. Eng. Dept., Aug. 1975.

A.3 Other Publications

I.A. Forbes, M.J. Driscoll, T.J. Thompson, I. Kaplan and D. D. Lanning

Design, Construction and Evaluation of a Facility for the Simulation of Fast Reactor Blankets

MIT-4105-2, MITNE-110, Feb. 1970

M.K. Sheaffer, M.J. Driscoll, and I Kaplan

A One-Group Method for Fast Reactor Calculations

MIT-4105-1, MITNE-108, Sept. 1970

A.3 Other Publications (continued)

I.A. Forbes, M.J. Driscoll, D.D. Lanning, I. Kaplan and
N.C. Rasmussen

LMFBR Blanket Physics Project Progress Report No.1

MIT-4105-3, MITNE-116, June 30, 1970

I.A. Forbes, M.J. Driscoll, T.J. Thompson, I. Kaplan and
D. D. Lanning

Design, Construction and Evaluation of an LMFBR
Blanket Test Facility

Trans. Am. Nucl. Soc., Vol.13, No.1, June 1970

S.T. Brewer, M.J. Driscoll and E.A. Mason

FBR Blanket Depletion Studies - Effect of Number of
Energy Groups

Trans. Am. Nucl. Soc., Vol. 13, No. 2, Nov. 1970

M.K. Sheaffer, M.J. Driscoll and I. Kaplan

A Simple One-Group Method for Fast Reactor Calculations

Trans. Am. Nucl. Soc., Vol. 14, No. 1, June 1971

T.C. Leung, M.J. Driscoll, I. Kaplan and D.D.Lanning

Measurements of Material Activation and Neutron
Spectra in an LMFBR Blanket Mockup

Trans. Am. Nucl. Soc., Vol. 14, No. 1, June 1971

S.T. Brewer, E.A. Mason and M.J. Driscoll

On the Economic Potential of FBR Blankets

Trans. Am. Nucl. Soc., Vol. 14, No. 1, June 1971

I.A. Forbes, M.J. Driscoll, N.C. Rasmussen, D.D. Lanning
and I. Kaplan

LMFBR Blanket Physics Project Progress Report No. 2

COO-3060-5, MITNE-131, June 1971

C.P. Tzanos, E.P. Gyftopoulos and M.J. Driscoll

Optimization of Material Distributions in Fast
Breeder Reactions

MIT-4105-6, MITNE-128, August 1971

A.3 Other Publications (continued)

T.C. Leung and M.J. Driscoll

A Simple Foil Method for LMFBR Spectrum Determination

Trans. Am. Nucl. Soc., Vol. 14, No. 2, Oct. 1971

C.S. Kang, N.C. Rasmussen and M.J. Driscoll

Use of Gamma Spectroscopy for Neutronic Analysis
of LMFBR Blankets

COO-3060-2, MITNE-130, Nov. 1971

T.C. Leung, M.J. Driscoll, I. Kaplan and D.D. Lanning

Neutronics of an LMFBR Blanket Mockup

COO-3060-1, MITNE-127, Jan. 1972

N.R. Ortiz, I.C. Rickard, M.J. Driscoll and N.C. Rasmussen

Instrumental Methods for Neutron Spectroscopy in the
MIT Blanket Test Facility

COO-3060-3, MITNE-129, May 1972

V.C. Rogers, I.A. Forbes and M.J. Driscoll

Heterogeneity Effects in the MIT-BTF Blanket No. 2

Trans. Am. Nucl. Soc., Vol. 15, No.1, June 1972

S.T. Brewer, E.A. Mason and M.J. Driscoll

The Economics of Fuel Depletion in Fast Breeder
Reactor Blankets

COO-3060-4, MITNE-123, Nov. 1972

M.K. Sheaffer, M.J. Driscoll and I. Kaplan

A One-Group Method for Fast Reactor Calculations

Nucl. Sci. Eng., Vol. 48, p. 459 (1972)

C.P. Tzanos, E.P. Gyftopoulos and M.J. Driscoll

Optimization of Material Distributions in Fast Reactor
Cores

Nucl. Sci. Eng., Vol. 52, p. 84 (1973)

A.3 Other Publications (continued)

M.J. Driscoll, D.D. Lanning and I. Kaplan

LMFBR Blanket Physics Project Progress Report No. 3

COO-3060-6, MITNE-143, June 1972

M.V. Gregory, M.J. Driscoll and D.D. Lanning

Heterogeneous Effects in Fast Breeder Reactors

COO-2250-1, MITNE-142, Jan. 1973

P.J. Wood and M.J. Driscoll

Assessment of Thorium Blankets for Fast Breeder Reactors

COO-2250-2, MITNE-148, July 1973

G.A. Ducat, M.J. Driscoll and N.E. Todreas

The Parfait Blanket Concept for Fast Breeder Reactors

Trans. Am. Nucl. Soc., Vol. 16, No. 1, June 1973

G.A. Ducat, M.J. Driscoll and N.E. Todreas

The Parfait Blanket Concept for Fast Breeder Reactors

COO-2250-5, MITNE-157, Jan. 1974

G.J. Brown and M.J. Driscoll

Evaluation of High-Performance LMFBR Blanket Configurations

COO-2250-4, MITNE-150, May 1974

P.A. Scheinert and M.J. Driscoll

Gamma Heating Measurements in Fast Breeder Reactor Blankets

COO-2250-10, MITNE-164, Aug. 1974

A.3 Other Publications (continued)

P.J. Wood and M.J. Driscoll

Economic Characterization of Breeder Reactor
Blanket Performance

Trans. Am. Nucl. Soc., Vol. 17, Nov. 1973

P.J. Wood and M.J. Driscoll

The Economics of Thorium Blankets for Fast Breeder
Reactors

Trans. Am. Nucl. Soc., Vol. 17, Nov. 1973

M.V. Gregory, M.J. Driscoll, D.D. Lanning

Heterogeneous Effects on Reactivity of Fast Breeder
Reactors

Trans. Am. Nucl. Soc., Vol. 17, Nov. 1973

M.J. Driscoll, et al.

LMFBR Blanket Physics Project, Progress Report
No. 4.

COO-2250-3, MITNE-149, June 30, 1973

M.J. Driscoll, et al.

LMFBR Blanket Physics Project Progress Report No. 5

COO-2250-14, MITNE-170, June 30, 1974.

A. Tagishi and M.J. Driscoll

The Effect of Reactor Size on the Breeding Economics
of LMFBR Blankets

COO-2250-13, MITNE-168, Feb. 1975

M.S. Kalra and M.J. Driscoll

Gamma Heating in LMFBR Media

COO-2250-18, MITNE-179, Feb. 1976 (est.)

A. Tagishi and M.J. Driscoll

The Effect of Core Size on Fast Reactor Blanket
Performance

TANSAO Vol. 21, June 1975.