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M.I.T. LMFBR BLANKET RESEARCH PROJECT FINAL SUMMARY REPORT

by

M. J. DRISCOLL

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U.S. Department of Energy Contract No.
DE-AC02-76ET37241.A004
(DOE/ET/37241-54; MITNE-257)

August 1983

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M.I.T. LMFBR BLANKET RESEARCH PROJECT

FINAL SUMMARY REPORT

by

M. J. DRISCOLL

AUGUST 1983

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Prepared for the U.S. Department of Energy under
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His colleagues on the faculty also played major roles which are much appreciated: the late Theos J. Thompson provided both encouragement and venture capital, and many others provided counsel and helped supervise thesis research: N. C. Rasmussen, D. D. Lanning, I. Kaplan and F. M. Clikeman, in particular, and also those listed as co-supervisors and co-authors in Sections 3 and 4.

Albert T. Supple, Jr. provided yeoman service as the project's Engineering Assistant in an amazing variety of roles; and Virginia Miethe and Rachel Morton were of invaluable assistance with our computer related endeavors.

Special recognition is also due the project's technical monitors at the AEC/ERDA/DOE: most recently, and of longest tenure, Philip B. Hemmig and John W. Lewellen; and in the early days, W. H. Hannum.

Distribution

MIT LMFBR Blanket Research Project

Final Summary Report

Department of Energy Research and Development

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M.I.T. LMFBR Blanket Research Project

Final Summary Report

by

M. J. Driscoll

ABSTRACT

This is a final summary report on an experimental and analytical program for the investigation of LMFBR blanket characteristics carried out at M.I.T. in the period 1969-1983. During this span of time, work was carried out on a wide range of subtasks, ranging from neutronic and photonic measurements in mockups of blankets using the Blanket Test Facility at the M.I.T. Research Reactor, to analytic/numerical investigations of blanket design and economics.

The main function of this report is to serve as a resource document which will permit ready reference to the more detailed topical reports and theses issued over the years on the various aspects of project activities. In addition, one aspect of work completed during the final year of the project, on doubly -heterogeneous blanket configurations, is documented for the record.

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SECTION 1

INTRODUCTION

1.1 Foreword

This is the final summary report of the LMFBR Blanket Project. It surveys the entire history of the project, from early 1968 through late 1983. During almost all of this period the project was supported by a succession of AEC/ERDA/DOE research contracts.

Although the project's work has been thoroughly documented in a succession of quarterly progress reports, which were concluded with the October-December 1982 issue, seven annual progress reports covering the period 1970 through 1976, and numerous other publications (see Section 3 of this report), it was felt that an overall summary publication would prove useful. In addition, the final phase of the FY '83 effort deserved to be recorded in a readily retrievable document.

1.2 Historical Perspective

Table 1.1 lists some key milestones in the life of the Blanket Research Project. As noted, construction of the Blanket Test Facility at the MIT Research Reactor began in July 1968, funded by an MIT research grant, as a doctoral thesis project by Ian A. Forbes. In July 1969, research support by the AEC commenced, and thus essentially all of the work actually completed in this area has been under AEC/ERDA/DOE auspices. Quite understandably, then, the program has been closely aligned to the overall U.S. LMFBR base technology effort, and the roster of research topics addressed over the years has reflected the evolving focus of the DOE program to an appreciable extent.

A parallel goal, in addition to advancing the state of the art in

Table 1.1

MIT Blanket Research Project Milestones

April 1967:	Internal Proposal (Thesis Prospectus).
July 1968:	Construction start on Blanket Test Facility (BTF) at MITR; cost is \$29,000 from Reactor Renovation Funds.
July 1969:	First AEC 3-year contract.
October 1969:	Start runs on First Blanket Mockup - a Steel and Borax Facility Test Assembly.
March 1970:	Start runs on Mockup No.2 - simulating a 1,000 MWe LMFBR.
February 1972:	Start runs on Mockup No.3 - a Graphite-Reflected Blanket.
March 1972:	Start of analytic work on Parfait (Internal Axial) Blanket concept.
January 1973:	Start runs on Mockup No.4 - a "Demo" reactor blanket (harder driver spectrum than No.2).
May 1974:	MIT reactor shutdown for renovations; Blanket Project focus becomes more analytical/numerical.
October 1976:	MITR-II resumes full power operations - BTF. neutron flux one-third that of MITR-I; cost of irradiation 3X higher. Mockup No.5 studied - one row of fuel with thick steel reflector.
January 1978:	Mockup No.6 in place: a 3-row blanket containing special assemblies: Thorium and Interface.
September 1978:	Experimental facility placed in standby status. Second phase of primarily analytical work.
December 1982:	Nominal end of project - return of fuel to DOE underway.

science and technology, has been the training of students who would go on to participate in the U.S. nuclear program. Since this aspect has not been highlighted in the past, a bit more detail is included here. Table 1.2 is a roster of students who have been associated with the project in one role or another, in roughly chronological order. To summarise the relevant statistics: some 40 students were aided by support as research assistants; 19 completed doctoral theses on the project, while 23, 3 and 4 did master's, engineer's and bachelor's thesis projects, respectively. In addition, another 31 students were associated to a lesser extent - carrying out special project research for subject credit, working part-time as lab assistants, etc.

Table 1.2

Roster of Students Associated with the MIT Blanket Project

1968-1983

Doctoral Students

I. A. Forbes
S. T. Brewer
J. N. Donohew
C. S. Kang
T. C. Leung
N. R. Ortiz
M. K. Sheaffer
C. P. Tzanos
A. Tagishi
G. A. Ducat
G. J. Brown
M. V. Gregory
M. S. Kalra
O. K. Kadiroglu
P. J. Wood
J. I. Shin
A. Salehi
M. Saidi
B. Atefi

Master's Students

C. W. Forsberg
S. L. Ho
A. Pant
P. G. Mertens
N. A. Passman
D. Shupe
W. J. Westlake
G. J. Brown
S. Y. Ho
J. K. Chan
T. P. Choong
R. J. Kennerley
R. E. Masterson
M. K. Yeung
D. A. Bruyer
C. A. Chambers
M. Ketabi
R. A. Morneau
R. A. Finnock
M. S. Macher
D. Aldrich
S. Ahmad
M. Plys

Engineer's Degree Students

P. A. Scheinert
S. S. Wu
H. Aminfar

Bachelor's Degree Students

D. Lal
A. M. Thompson
F. R. Field
D. W. Medeiros

Table 1.2 (continued)

Research Assistants*

I. A. Forbes
 S. Ahmed Ali
 S. T. Brewer
 D. K. Choi
 J. N. Donohew
 C. S. Kang
 T. C. Leung
 E. L. McFarland
 G. J. Brown
 P. L. Hendrick
 C. P. Tzanos
 P. Delaquil
 S. Y. Ho
 J. K. Chan
 T. P. Choong
 G. A. Ducat
 M. V. Gregory
 O. K. Kadiroglu
 P. A. Scheinert
 A. M. Thompson
 M. S. Kalra
 Y. Lukic
 J. I. Shin
 A. Tagishi
 R. A. Morneau
 R. A. Pinnock
 C. Chambers
 L. Metcalfe
 A. Salehi
 S. Keyvan
 M. Macher
 S. Kim
 D. Lancaster
 B. Atefi
 A. Kamal
 M. Plys
 S. Ahmad
 W. T. Loh
 A. Wolford
 F. Yarman

Other**

W. R. Corcoran
 J. L. Klucar
 P. Mockapetris
 J. W. Synan
 I. C. Rickard
 V. C. Rogers
 L. T. Kim
 J. L. Lazewatsky
 K. D. Roberson
 A. Alvim
 C. M. Hove
 L. Lederman
 A. S. Leveckis
 J. Pasztor
 E. Falco
 O. Basaran
 A. El-Magboub
 D. Wargo
 H. Khan
 T. Reckart
 F. Mobin
 W. Wolfe
 L. Paglia
 G. Nakayama
 W. J. Glantschnig
 S. W. Kim
 J. A. Sefcik
 R. Rogus
 E. Catron
 C. G. Heinzl
 A. Wolford

* Does not distinguish between full vs part-time.

** Fellowship/self-supporting/special project/laboratory assistant, etc.

1.3 Summary Description of the BTF

Although a detailed description of the Blanket Test Facility is available in the report by Forbes et al. (see Section 3, this report), a brief description follows because of the relevance of the subject matter to an understanding of the rationale underlying the entire research project.

The heart of the Blanket Test Facility is a uranium-loaded converter plate. Powered by the thermal neutron flux from the MITR thermal column, the converter generates the fast neutron flux for testing a mockup of an LMFBR blanket. The highly-thermalized flux provided by the thermal column (cadmium ratio for gold ≈ 1700) helps ensure that the converter plate has the leakage spectrum and albedo characteristic of the core of a large LMFBR.

The Blanket Test Facility (BTF) is located at the rear of the graphite-lined cavity, or "hohlraum", of the MITR; Figures 1.1 and 1.2 show cross section and plan views, respectively, of the facility.

The BTF irradiation region was formed by removing the rear graphite wall of the hohlraum and existing thermal exponential facility shielding, and installing a five-sided aluminum box to line the resulting cavity. The interior volume of this cavity is about 6 ft by 6 ft by 6 ft. The aluminum liner box is seam-welded to maintain the sealed nature of the hohlraum and prevent the leakage of A^{41} from the hohlraum into the irradiation region and the reactor building. Boral sheet is attached to the top, sides and floor of the box to reduce back-scattering of thermalized neutrons from the surrounding concrete shielding.

Shielding for the irradiation region is provided by four heavy concrete shield blocks, two stationary and two portable. The two port-

Fig. 1.1 Schematic Cross Section View of Hohlraum and Blanket Test Facility.

Fig. 1.2 Schematic Plan View of the Hohlraum and Blanket Test Facility.

able blocks weigh approximately 15 tons each and can be removed with the reactor building's overhead crane to provide access to the irradiation region.

A set of rails, extending from the front of the irradiation region out to the reactor building containment wall, permits the insertion of the cart-mounted experimental assemblies into the irradiation region.

For operation of the Blanket Test Facility, the converter assembly is rolled to the front of the irradiation region, the simulated blanket assembly is installed directly behind it, and the concrete shield doors replaced. For operation of the thermal exponential facility and other experiments utilizing the hohlraum thermal flux, the converter assembly is replaced by a third cart, loaded with graphite and having the same dimensions as the converter assembly. The graphite cart restores the reflective properties of the hohlraum for thermal neutrons.

Changeovers from fast operation to thermal operation and vice versa are performed only when the MIT Reactor is shut down. However, personnel dose rates are low enough to permit access to the BTF irradiation region when the reactor is at full power (5 Mw), for the retrieval of experimental packages (foils, etc.) from the blanket assembly.

A schematic view of the converter assembly is shown in Fig. 1.3. The converter assembly consists of a graphite external moderator region composed of 4 in. by 4 in. reactor grade graphite stringers, and a fuel region composed of 1/2-in.-diameter, aluminum-clad UO_2 fuel rods in a close-packed, triangular-pitch array. The active fuel region is 48 in. high and 60 in. wide, and has U^{235} enrichments of 1.0999% and 1.99%. The graphite stringers and fuel rods are mounted vertically, between upper and lower grid plates, in a closed aluminum container. The

Fig. 1.3 Cutaway View of the Converter Assembly.

container is designed to prevent any accidental rearrangement of fuel that might approach a critical assembly even if the facility were flooded with H_2O ; it will also contain any A^{41} generated in the converter.

In the initial converter assembly loading, the graphite external moderator region was 8 in. thick and the UO_2 fuel region 6.865 in. thick. However, both the graphite and fuel loadings may be readily changed (provision is made for up to 12 in. of graphite and 7.755 in. of fuel) to permit the generation of a wide range of converter leakage spectra, and the graphite region was in fact reduced in thickness part way through the experimental program to harden the spectrum incident on the blankets under test - corresponding to a switch from a commercial to a demo size core simulation.

BTF Blanket No.2, the first realistic assembly tested, and the prototypic unit for all that followed, was an accurate mockup of a typical LMFBR blanket composition. Subassembly boxes of low-carbon steel rectangular mechanical steel tubing were loaded with 121 uranium metal fuel rods arranged on a square lattice spacing of 0.511 in.; the 0.25-in. diameter uranium metal fuel is clad in low-carbon steel tubing. The inter-rod volume in each subassembly was filled with anhydrous sodium chromate (Na_2CrO_4) powder. The subassembly boxes were loaded on an experimental cart to provide a blanket assembly 48 in. high, 59.2 in. wide and 17.72 in. thick. The blanket was backed by an 18-in. thick low-carbon steel reflector.

The as-loaded atom densities for Blanket No.2 are given in Table 1.3.

Figure 1.4 shows a schematic view of Blanket Assembly No.2. A $58\frac{1}{4}$ -in. by $62\frac{7}{16}$ -in. piece of one inch thick mild steel plate welded

TABLE 1.3

Homogenized Atom Densities in B. T. F. Blanket No. 2

Nuclide	Blanket No. 2	Equivalent Realistic Blanket*
U ²³⁵	0.000088	0.000016
U ²³⁸	0.008108	0.008131
O	0.016293	0.016293
Na	0.008128	0.008128
Cr	0.004064	0.003728
Fe	0.013750	0.012611
Ni	0.000000	0.001475
H	0.000073	0.000000
C	0.000096	0.000082

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

Fig. 1.4 Schematic View of Blanket Assembly No. 2.

between two 60 in. by 39 in. pieces of one inch thick mild steel plate forms an "H" frame support structure which is mounted on an experimental cart. The front section of the "H" frame contains three rows of the blanket subassemblies, and the rear section is filled with seventeen $58\frac{1}{4}$ -in. by 60 in. pieces of one inch thick mild steel plate to act as a neutron reflector. Twenty-five of the subassemblies contain steel-clad uranium metal fuel rods and anhydrous sodium chromate powder. The outer subassemblies (see Fig. 1.4) are filled with the mixture of iron punchings and anhydrous borax ($\text{Na}_2\text{B}_4\text{O}_7$) powder used for Blanket Assembly No. 1 (an ersatz blanket used only for facility checkout purposes).

Twenty-six tubes are provided for foil activation traverses in the axial and transverse directions through the blanket (see Fig. 1.5). The 58-in. long mild steel tubes have a 7/16 in. o.d. and a 0.028 in. wall thickness. A one inch diameter hole, 4 inches below midplane, has been drilled through the reflector to provide a beam hole for fast neutron and prompt gamma spectrum measurements. In addition, a foil holder rod may be inserted in this hole for foil activation traverses through the reflector region.

Figure 1.6 shows a schematic view of a typical subassembly. The low-carbon steel subassembly boxes are 5.92 inches square, 60 in. high and have a wall thickness of approximately 3/32 in. The bottom of each subassembly is sealed with a seam-welded steel plate. Each subassembly contains 121 fuel rods arranged in an eleven by eleven square lattice with a pitch of 0.511 in. Sixty of the rods have a U^{235} enrichment of 1.016%, and sixty-one have a U^{235} enrichment of 1.143%; the two enrichments are loaded in a checkerboard pattern within the

Fig. 1.5 Plan View of Blanket Assembly Showing the Traversing Tube Positions.

Fig. 1.6 Schematic Cross Section View of Blanket No. 2 Subassembly.

subassembly box.

The fuel rods are held in place by upper and lower aluminium grid plates. The lower grid plate rests on the bottom closure plate, and the upper grid plate is supported on four 48-in. long tubes which have an o. d. of 7/16-in. and a wall thickness of 0.028 in. These tubes fit over four fuel rods located near the corners of the lattice. The fuel rods are loaded through the upper grid down into the lower grid plate. The upper grid plate has cut-out sections for the traversing tubes (see Fig. 1.5) and for loading the sodium chromate powder; each tube normally contains a fuel rod unless a foil traverse is to be made.

A total of 3025 fuel rods were fabricated at MIT by recladding 48-in. long by 0.250-in. diameter uranium metal rods in low-carbon steel tubing. The clad tubing is 50 in. long, and has a 5.16-in. o. d. and a 0.018-in. wall thickness. Each end of the tube is closed by a press-fitted steel plug, 1/2-in. long and 9/32-in. in diameter.

The inter-rod volume of the subassemblies (see Fig. 1.7) is filled with anhydrous sodium chromate powder (technical grade) which has been dried and ground. The average loading of sodium chromate in a subassembly is 31.106 kg, with a standard deviation of ± 0.294 kg; the loadings vary from 30.51 kg to 31.80 kg, or $\pm 2\%$ maximum deviation from the mean.

The top of each subassembly is sealed by a 0.035-in. thick steel plate which is epoxied in place to ensure that the subassembly is air- and water-tight; the traversing tubes penetrate this plate.

A breakdown of the subassembly weight is given in Table 1.4.

The atom densities for Blanket No.2 were calculated by homogenizing the material components of a subassembly at mid-height, viz. the

Fig. 1.7 Blanket No. 2 Unit Cell.

TABLE 1.4
Subassembly Component Weights

Uranium metal	89.30 kg
Na_2CrO_4	31.11 kg
Cladding	13.00 kg
Subassembly box	26.55 kg
Grid plate support tubes	0.91 kg
Grid plates	0.36 kg
Total	161.23 kg

uranium metal fuel, the anhydrous sodium chromate and the low-carbon steel cladding, support tubes and subassembly walls. The carbon content of the steel is about 0.15 w/o; other impurities, such as manganese and nickel, are negligible. The water content of the sodium chromate is 0.10 w/o.

The homogenized atom densities in Blanket No.2 are given in Table 1.3 where they are compared with the atom densities in an "equivalent realistic blanket", composed of 37.0 v/o depleted UO_2 (at 90% of theoretical density), 20.7 v/o Type 316 stainless steel (71.2 w/o Fe, 20.0 w/o Cr and 8.8 w/o Ni), 32 v/o sodium and 10.3 v/o void. It is evident that Blanket No.2 provides a realistic blanket composition in all respects, with the exception of the small hydrogen content.

Blankets of this general type, and variations thereon, as noted in Table 1.1, served as the subjects of study in and of themselves, or as host assemblies for other experiments over the eight or so years in which an active experimental program was in force.

While it is difficult to classify the rather eclectic menu of research topics pursued over the long history of the project, this task is essayed in Table 1.5. Other topic matrices could be used to rearrange the same roster of subtasks. It is beyond the scope of this endeavor to cite the specific contributions under each category - for this the reader is referred to Section 4 of this report, which records, verbatim, the abstracts of all major reports and theses - nearly 50 in all. Special attention is called to the report by Macher, who compiled and analyzed most of the generic data on experimental blanket mockup studies (as opposed to specific studies of methodologies, paper studies, etc.).

TABLE 1.5

SUMMARY OF MAJOR RESEARCH AREAS

Experimental*

Design of Converter-Driven Blanket Facility
Measurement of Breeding-related Characteristics
Foil Neutron Spectrometry
Instrumental Neutron Spectroscopy
Gamma Spectroscopy, Prompt and Decay
Reflected Blanket Characteristics
Reflector Neutronics and Photonics
Thorium Assembly Breeding Performance
Gamma Heating and Dosimetry
Heterogeneous Self-Shielding

Analytical/Numerical*

One-Group Method for FBR Calculations
Fuel Pin Heterogeneity
Interface Heterogeneity
Optimization of Blanket Composition and Configuration
Parfait and Related Core/Blanket Arrangements
Thorium and Moderated Blankets
Blanket Fuel Management
Blanket Economics
Thermal-Hydraulic Performance of Special Blanket Assemblies
Selected Aspects of Cross Section Generation
Benchmark Problem Calculations

*Note that many studies, in fact, combined both aspects.

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SECTION 2

SUMMARY OF FY '83 EFFORT

2.1 Foreword

The work in question here took place mainly during calendar 1982. First of all, the project contributed (within its computational capabilities) to the Large Core Code Evaluation Working Group (LCCEWG) Benchmark Problem No. 4. Results were essentially confined to R-Z computations. They will not be repeated here since full documentation and a comparison among all participants will be undertaken by the LCCEWG itself.

A second thematic activity is the focus of the discussion here - an investigation into the potential benefits and drawbacks of multiply-heterogeneous LMFBR cores, that is, cores having both radial and axial internal blankets. The full text of a report submitted on this topic by A. J. Wolford is reproduced in full in the following pages of this section, since it has not been published elsewhere in one of the project's several modes of reporting. The "bottom line" on this area of research is apparently that a "highly heterogeneous" system - one with many (necessarily) small subdivisions of interspersed fertile and driver fuel - rapidly approaches the characteristics of a homogeneous system: i. e. the long neutron mean free path in a fast reactor begins to re-homogenize the subregions. Hence the advantages of heterogeneity are lost, and one is advised to limit heterogeneous designs to fairly large blanket and driver subregions.

2.2 Introduction

It is widely known that a heterogeneous core configuration is an effective means of reducing sodium void reactivity in large LMFBRs. This is physically due to the increase in neutron leakage to fertile blanket regions (with respect to a comparable homogeneous core) when boiling occurs in the driver fuel region.

There are many configurations to consider when optimizing the arrangement of blanket assemblies within a core, the two most notable being axially heterogeneous and radially heterogeneous designs. Radial blanket designs (bullseye cores) have been analyzed in exhaustive detail [I-1, S-1]; but there has been significantly less interest in LMFBR core arrangements containing axial internal blankets.

Optimization of axial blanket thickness has been studied by previous workers at MIT [L-1, D-1], and a typical value of 30 cm total axial dimensions is suggested. It has also been shown by this work that a single central axial internal blanket is preferred to the use of several axial internal layers.

There is reason to expect that a combination of these two heterogeneous concepts merits further investigation. The following work undertakes this task by evaluating radial and axial heterogeneities distinctly and in combination.

2.3 Core Modelling

A. Theoretical Basis

The positive void coefficient of reactivity in an LMFBR is actually the net sum of two rather large opposing physical phenomena; the

reactivity loss due to neutron leakage and the reactivity gain due to spectral hardening and consequent increase in eta. There are two other less significant effects generally considered to be negligible by comparison [W-1]; the change in self-shielding and the reduction in sodium capture.

The heterogeneous core concept takes advantage of the increased leakage to fertile blanket regions to reduce the overall positive effect. It is evident that this leakage effect must be strongly space-dependent, as leakage varies with the flux gradient, which, in turn, varies throughout the core. It is also reasonable to expect that this leakage reactivity loss will correlate strongly with the driver-blanket leakage surface.

B. Cell Model

In developing a model for comparing the interesting cases of combined heterogeneity, a major concern was to employ as simple a configuration as possible so as not to obscure fundamental results. This conclusion was reinforced by the results of previous work at MIT [M-1], in which calculations were performed on fully-detailed, two-dimensional geometries.

The modelling approach taken in this work was to define a cylindrical "cell" simulating the first two regions of a central blanket bullseye core (or, equivalently, one module of a large hexagonal-pattern modular core). Representative dimensions and compositions were taken from the Large Core Code Evaluation Working Group Benchmark Problem 4 [L-3]. Sixty-degree sectors (R, THETA) are compared in Fig. 2.4 for the actual blanket and driver and the equivalent cell.

Since the intent of this research is to evaluate two dissimilar types of heterogeneity in combination, the coupling effects of multiple

radial driver regions were excluded from the problem by imposing an albedo of unity (reflective boundary condition) on the outer radial cell surface. Leakage was treated differently in the axial direction, however, as constant thickness fertile blankets were retained at the cell axial extremities.

All sodium void reactivity savings were compared to a reference cell composed of three axial zones; a homogenized core region reflected at the top and bottom by fertile axial blankets. This reference cell was designated REFMOD, and subsequent cells were designated NAMOD1 through NAMOD5. Single radial (NAMOD1) and axial (NAMOD2) heterogeneities were modelled as depicted in Fig. 2.2, and combined (NAMOD3) as in Fig. 2.1. In all cells, the total mass of plutonium was conserved in the driver regions.

A variant of the doubly-heterogeneous cell was considered by staggering the position of the axial internal blanket in the driver with respect to the blanket (NAMOD4), and is depicted along with its complement (NAMOD5) in Fig. 2.3. The intent of the staggering is to retain the advantage of axial lumping, while increasing the driver-blanket leakage surface.

Each of these cells will be investigated by performing a standard eigenvalue search. Sodium void reactivity savings will be the primary evaluation criterion, but other performance characteristics will also be surveyed.

C. Analytical Methods

A description of the methods used to solve the eigenvalue problem in multigroup form and to determine the energy group constants follows.

Fig. 2.1 Doubly-Heterogeneous and Reference Core Configurations.

Fig. 2.2 Radially and Axially Heterogeneous Core Configurations.

Fig. 2.3 Staggered Doubly-Heterogeneous Core Configurations.

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Fig. 2.4 Sixty-Degree Sectors.

1. The 2DB Code

The code used in performing the eigenvalue searches for all cells was revision one of Battelle Northwest's (original) 2DB by Little and Hardie [L-4]. Salient features of the code are briefly described below.

2DB is a two-dimensional multigroup diffusion code applicable only to fast reactors. The spatial flux distribution is solved by mesh-centered finite-difference equations, and energy detail is limited to fifty (50) or fewer groups. Flux profiles are computed by the familiar source-iteration technique. For each outer iteration (e.g. computing flux distributions for all groups, then re-calculating a new fission source) a multiplication ratio is defined by the ratio of new to old fission sources. By normalizing the source from each outer iteration by the multiplication ratio from the previous outer iteration, the eigenvalue, λ , is taken as the cumulative product of all multiplication ratios. In this way the eigenvalue approaches unity as the iterations proceed. Convergence is simply defined as $|1-\lambda| < e$, where e is user-specified.

Although 2DB contains a full capability burnup algorithm, the basis of comparison for the purposes of this work is at cell BCL, and hence this capability of the code is unused.

2. Energy Group Constants

The group constants employed in this study were taken from the LCCEWG Benchmark Problem 4 [L-3]. The library has an 8-group energy structure and was generated at Hanford Engineering Development Laboratory by LCCEWG participant R. E. Schenter. The library was processed from ENDF/B-V data. Cross sections for all isotopes are self-shielded by explicit modelling of the pin and lattice geometry. Bucklings employed were obtained from the LCCEWG Benchmark

Problem 3 to generate flux solutions for use in collapsing from 70 to 8 groups. Axial end blanket material was assumed to be at a temperature of 900 K, and driver and internal blanket material at 1500 K.

D. Static Calculations

A static eigenvalue search was performed for each cell: REFMOD and NAMOD1 through NAMOD5. The number densities in all cases simulate a BOL core, hence no fission product inventory was included. The results of these static calculations are tabulated in Table 2.1.

To determine the sodium void reactivity, 20% of the sodium mass was homogeneously removed from all zones except the axial and central radial blankets. Differential results of the voided cell calculations also appear in Table 2.1.

E. Results and Analysis

As previously stated, the results of all static calculations are compiled in Table 2.1. Indeed, it is evident that at 0.174% k/k for a 20% homogeneous voiding, NAMOD3 yields superior results and REFMOD, alternately, has the greatest positive worth. NAMOD1 and 2 results both fall within the above limiting cases, and this group of four "simple" combinations indicate that reactivity correlates directly with the total driver-blanket surface S_{bd} .

This simple correspondence fails to predict relative (to reference case REFMOD) void reactivity for the case of more complex geometries as is the case with the staggered cells NAMOD4 and 5. A more detailed analysis is required to explain these phenomena.

It can be motivated in an informal manner (see Appendix A) that substantial increases in the driver-blanket interface are obtained at the expense of preserving distinct driver and blanket regions. In other

CORE	REFMOD	NAMOD1	NAMOD2	NAMOD3	NAMOD4	NAMOD5
k_{eff}	1.1251	1.2001	1.0873	1.1806	1.1892	1.1218
$\Delta k/k(\%)$	0.5523	0.3384	0.3110	0.1739	0.4962	0.4186
S_M	1.42×4	2.41×4	2.83×4	3.03×4	2.57×4	3.11×4
V_d	8.54×5	5.96×5	6.48×5	4.47×5	6.48×5	6.48×5
V_b	5.04×5	7.71×5	7.20×5	9.20×5	7.20×5	7.20×5
$\bar{l}_d$	243.84	98.99	91.45	59.05	100.80	83.33
$\bar{l}_b$	142.24	128.02	101.59	121.49	111.99	92.58

Table 2.1 Reactivity Values and Geometric Parameters.

	REFMOD	NAMOD1	NAMOD2	NAMOD3	NAMOD4	NAMOD5
$\hat{P}/\bar{P}$ (Driver)	1.309	1.422	1.214	1.026	1.345	1.535
$\hat{P}/\bar{P}$ (Core)	2.393	3.264	3.715	2.292	2.838	3.241
P_b/P_d *	0.	0.035	0.067	0.090	0.113	0.105

Cell Power Characteristics-20% Na Void

	REFMOD	NAMOD1	NAMOD2	NAMOD3	NAMOD4	NAMOD5
$\hat{P}/\bar{P}$ (Driver)	----	1.430	1.222	1.090	1.411	1.549
$\hat{P}/\bar{P}$ (Core)	----	3.282	3.738	2.302	2.978	3.270
P_b/P_d *	----	0.034	0.065	0.088	0.106	0.102

Cell Power Characteristics-Unvoided

Table 2.2 Comparison of Power Shaping Characteristics.

* Blanket/Driver Power Ratio.

words, on a scale of ~ 30 cm the advantage of lumping fuel and blanket material begins to decrease quite significantly, and the heterogeneous core begins to appear as a homogenized material to the neutron population. These general results are confirmed by the findings of W. P. Barthold et al. [W-1], wherein it is concluded that the reactivity of internal blanket assemblies is greater for tightly coupled (neutronically) cores than loosely coupled core configurations.

An evaluation of power peaking effects also supports the advantage of the non-staggered doubly-heterogeneous cells. Table 2.2 is a compilation of both voided and unvoided power characteristics for all cells investigated. NAMOD3 has a considerable (25%) advantage in power peak reduction over NAMOD4, the better of the two staggered cells. It should be stressed that all cells produce the same thermal power even though neither blanket nor driver volumes are conserved individually (the total cell volume is the same for all cells).

2.4 Summary

A. Conclusions

The preceding investigation of combined heterogeneity in LMFBR core design indicates that the driver-to-blanket leakage surface is a controlling parameter in reducing sodium void reactivity worth. It has also been shown that selection of the optimum leakage surface is not a simple maximization process. Careful attention must be given to increasing this driver-blanket interface in staggered doubly-heterogeneous cores, as a homogenization effect can occur in driver and blanket lumps of the order of ~ 30 cm.

Staggered doubly-heterogeneous cells do not exhibit a distinct advantage over the non-staggered versions, in terms of power peaking or

sodium void reactivity worth reduction. The combined radial and axial internal blanket cell, on the other hand, clearly performs better than either of its constituent members individually.

B. Recommendations

A similar study comparing several staggered and non-staggered cells holding driver volumes constant would aid in determining the precise dependence of void reactivity on driver-blanket leakage surface. Other analyses further investigating optimum lump dimension or surface to volume ratio would also expand the implications of this work and method.

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APPENDIX A

It is the intent of the text that follows to informally motivate the homogenized behavior of the multiply-heterogeneous, staggered-blanket cell configurations NAMOD4 and NAMOD5.

It is well-known that the escape probability from an isolated fuel rod (surrounded by an infinite sea of moderator) is given by the Wigner Rational Approximation:

$$P = \frac{1}{1 + \Sigma_{\text{tfuel}} \bar{\ell}_f}$$

in which $\bar{\ell}_f$ is the Dirac mean cord length in the fuel.

$$\bar{\ell}_f \equiv \frac{4 V_f}{A_f}$$

If we employ an analogous method (loosely, as the blanket region is not convex as is required for the approximation) for our isolated cells (recall the early imposition of a reflective boundary condition), the resulting expressions for the escape probabilities from blanket-to-driver, P_{bd} , and driver-to-blanket, P_{db} , are:

$$P_{bd} = \frac{1}{1 + \frac{4 \bar{\Sigma}_{tb} V_b}{A_{bd}}}$$

$$P_{db} = \frac{1}{1 + \frac{4 \bar{\Sigma}_{td} V_d}{A_d}}$$

Both of these expressions approach unity as the blanket driver interface is increased from some comparable reference case. This conclusion is, in fact, stating the homogeneous limit, the only condition which allows both probabilities to be unity simultaneously.

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SECTION 3

BIBLIOGRAPHY OF PUBLICATIONS

In this section all publications associated with work performed on the Blanket-Physics Project are listed in approximately historical order. Doctoral and Engineer's Theses are listed first, followed by S.M. and S.B. Theses, and then by other publications.

Abstracts for all theses are reproduced in Section 4; note that all doctoral theses are also published as topical reports.

Copies of all theses can be obtained from MIT through normal channels (see Section 4). Reports are available via NTIS.

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SECTION 4

ABSTRACTS OF MAJOR PUBLICATIONS

The pages which follow reproduce, in approximately historical order, the abstracts from all of the major topical reports and theses published under project auspices. All doctoral theses are also published as topical reports; the report versions are reproduced here.

Copies of theses may be obtained from MIT via normal channels; most of the reports are available from NTIS in the same manner as for all other AEC/ERDA/DOE sponsored work elsewhere.

We have not reproduced the other publications cited in Section 3; most are in readily available literature.

Copies of the current (July 1983) order forms from the MIT Microreproduction Laboratory are appended. They should be contacted for up-to-date prices and procedures.

MIT-4105-2

MITNE-110

February, 1970

DESIGN, CONSTRUCTION AND EVALUATION
OF A FACILITY FOR THE SIMULATION
OF FAST REACTOR BLANKETS

by

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

ABSTRACT

A facility has been designed and constructed at the MIT Reactor for the experimental investigation of typical LMFBR breeding blankets. A large converter assembly, consisting of a 20-cm-thick layer of graphite followed by a 17.5-cm-thick UO_2 fuel region, is used to convert thermal neutrons into fast neutrons to drive a blanket mockup. Operating at 55 watts, the converter generates blanket fluxes at an equivalent LMFBR core power of about 350 watts, with as little as one-tenth of the blanket material required for a critical assembly. Calculations show that the converter leakage spectrum is a close approximation to the core leakage spectrum from reference LMFBR designs, and that the axial distribution of the neutron flux in the blanket assembly simulates that in the radial blanket of a large LMFBR when the effective height and width of the blanket assembly are correctly chosen.

Testing of the completed facility with a blanket composed of 50 v/o iron and 50 v/o borax showed that the lateral flux distributions were cosine-shaped, and that lateral spectral equilibrium was achieved in a large central volume of the blanket. Backscattering from concrete shielding surrounding the experiment was found to affect no more than the outer 30 cm of the blanket assembly, confirming the results of two-dimensional multigroup calculations. Measurements of the axial activity of gold and indium show good agreement with 16-group, S_8 ANISN calculations.

MEASUREMENT OF FAST AND EPITHERMAL NEUTRON SPECTRA
USING FOIL ACTIVATION TECHNIQUES

by

Sung Ling Ho

Submitted to the Department of Nuclear Engineering
on January 15, 1970 in partial requirement for the Degree of
Master of Science.

ABSTRACT

A foil activation method was used to measure fast and epithermal neutron spectra. A set of 7 resonance foil detectors was used to measure the epithermal neutron spectrum on the surface of a one inch natural uranium rod in the MITR D₂O exponential facility, and a set of 7 threshold foils was used to measure the fission neutron spectrum in the MITR transistor irradiation facility. Unlike previous activation methods in which the foil materials are separated and counted individually, a one-step simultaneous counting procedure was developed based on Ge(Li) spectrometry. The GAMANL code was used to extract individual nuclide activities. These activities were then fed into the existing SAND II code to unravel the neutron energy spectrum. Analytical methods were also used to calculate the flux. The measured and calculated fluxes in the transistor irradiation facility were comparable. Owing to insufficient energy coverage, no acceptable solution for the epithermal spectrum on the surface of the single rod was obtained from SAND II.

Thesis Supervisor: M. J. Driscoll

Title: Professor of Nuclear Engineering

INELASTIC SCATTERING IN FAST REACTOR MATERIALS

by

JACK N. DONOHEW, JR.

Submitted to the Department of Nuclear Engineering on March 19, 1970, in partial fulfillment of the requirements for the degree of Doctor of Science.

ABSTRACT

Inelastic scattering models have been developed for fast reactor calculations to determine multi-group scattering removal cross sections, σ_R , for the resolved and the unresolved nuclear level regions. The computer model for the resolved region permits calculation of σ_R from its integral definition and basic cross section data. For the unresolved region, where individual levels lie too closely together to be experimentally resolved, the individual level model developed in this thesis assumes that the nuclear level structure can be represented by a series of individual, non-interacting levels. Sample calculations involving U-238, since this is the single most important inelastic scattering material in fast reactors, are presented and evaluated.

The conclusions arrived at were: that the "recipe" used to calculate elastic σ_R , for representative multi-group sets (YOM and ADBN sets), are valid; that the "recipe" used to calculate inelastic σ_R underestimates the energy loss predicted by the neutron slowing-down equations; and that the choice of the weighting flux spectrum was unimportant in the calculation of either σ_R for high A materials (i.e. $A \sim 238$). The inelastic decrements predicted for the U-238 unresolved region by the individual level model were compared to those predicted by the statistical model, the most commonly applied model in the preparation of multi-group sets. The results support, in a qualitative manner, the common assumption that the statistical model is valid to describe inelastic scattering in the unresolved region.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

ABSTRACT

AN EVALUATION OF A SUBCRITICAL NULL-REACTIVITY METHOD FOR
FAST REACTOR APPLICATIONS.

by

Paul Gustaaf Mertens

Submitted to the Department of Nuclear Engineering on May 20, 1970 in partial fulfillment of the requirements for the degree of Master of Science.

A new subcritical method for measuring the infinite medium multiplication factor, k_{∞} , of fast reactor materials has been evaluated. The method is closely related to the critical PCTR or null-reactivity technique and involves, after generation of the infinite reactorspectrum, the addition of poison in the testregion until a suitable "axial buckling" matching condition is reached.

Problems associated with the practical realisation of the concept in the MITR exponential facility were investigated. The design adopted consists of a testregion 10 cm in diameter, surrounded by an annular converter region 14 cm thick made up of a close-packed array of 2 % enriched fuel rods, which is in turn surrounded by a thermal region consisting of a carbon powder giving an overall system diameter of 72 cm.

A new method for adjusting the energy spectrum of the neutrons in the test region was investigated : namely, varying the density of the carbon in the thermal region, and was found to be quite effective. It was also found that the error in the axial buckling measurement which is the major experimental indication of the null condition can be made sufficiently small by optimization of the design.

The major problems remaining before practical realization of the concept is possible, were also outlined and approaches suggested for attacking these problem areas, the major one being the radial buckling mismatch.

Thesis Supervisor : Michael J. Driscoll
Title : Associate Professor of Nuclear Engineering

HETEROGENEOUS EFFECTS IN LMFBR BLANKET FUEL ELEMENTS

by

WILLIAM JAMES WESTLAKE, JR.

Submitted to the Department of Nuclear Engineering
on June 4, 1970 in partial fulfillment of the requirements
for the degree of Master of Science.

ABSTRACT

Measurements of the intra-rod U-238 fission and capture reaction rate distributions were made in 1/4-inch diameter, 1.016% U-235 enriched, uranium metal fuel rods. The fuel rods were irradiated in the M.I.T. LMFBR Blanket Test Facility, which provides a neutron flux having an energy spectrum characteristic of a LMFBR blanket. Three sets of uranium foil activation measurements were performed at successive depths into blanket assembly No. 2 of the BTF. The experimental results indicate that the heterogeneous effect on the U-238 capture rate within blanket fuel rods, as reflected in the ratio of rod surface to rod centerline activation, is appreciable (i.e., surface to centerline capture rate ratios ranging from 1.10 to 1.20 for various blanket depths). Consideration of this heterogeneity effect on experiments carried out in the BTF and ultimately in the design of LMFBR blankets is clearly warranted.

The experimental results were compared with one-dimensional, 16 energy group, ANISN, discrete ordinate computations and with predictions based on simplified analytical formulations. These comparisons indicated that the observed heterogeneity effect on intra-rod U-238 capture rates is due primarily to U-238 resonance self-shielding. Essentially no heterogeneity effect on U-238 fast fission in blanket fuel rods was obtained within experimental error (i.e., fission reaction rate data accurate to $\sim \pm 1\%$). These results do not, however, contradict the theoretical prediction that the observed U-238 fast fission heterogeneity effect would correspond to only a 0.4% rod-centerline flux-peak in the intra-rod U-238 fission reaction rate distribution.

A general discussion of the role of heterogeneous effects in both the design of LMFBR blankets and the evaluation of experimental data from fast critical assemblies is also included. Finally, the results of a parametric study of the U-238 fast fission heterogeneity effect in various LMFBR systems, for both the core and the blanket, based solely on the contribution of first flight neutrons in their unit cell of origin, are presented.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

THE FEASIBILITY OF INFERRING THE INCIDENT NEUTRON
SPECTRUM FROM PROMPT CAPTURE GAMMA-RAY SPECTRA

by

Dwight Ardan Shupe

Submitted to the Department of Physics on August 24, 1970,
in partial fulfillment of the requirements for the degree
of Master of Science.

Abstract

This thesis investigates an important problem currently limiting experimental reactor physics research pertinent to the development of fast breeder reactors: development of an effective neutron spectrometer for energies below 1 keV. In this work a prototype instrument was constructed to investigate experimental aspects of the feasibility of developing a neutron spectrometer based on in-situ, prompt capture gamma-ray analysis. Ta-181 was selected as the best target material for the spectrometer.

The equipment constructed for this research consisted primarily of a sophisticated collimating unit for extracting the Ta-181 prompt gamma beam to a high resolution Ge(Li) gamma-ray spectrometer. This apparatus was tested on both thermal and fission spectrum neutron fluxes, which bracket the energy range of the intended eventual applications.

The results of this work show that the prototype facility does produce and extract some 54, Ta-181 lines suitable for analysis, despite high background, low count rates, and attendant electronic problems such as severe peak drift. The work is being continued by others to assess the feasibility of developing analytical and numerical methods for inferring the incident neutron spectrum from the measured prompt capture gamma-ray spectra.

Thesis Supervisor: Michael J. Driscoll
Title: Assistant Professor of
Nuclear Engineering

MIT-4105-1
MITNE-108
September, 1970

A ONE-GROUP METHOD FOR FAST REACTOR CALCULATIONS

by

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

ABSTRACT

A one-group method for calculation of neutron balances in fast breeder reactor cores is developed and evaluated. The key feature of the method is the definition of two spectrum characterization parameters,

$$S = \frac{\nu \Sigma_f}{\nu \Sigma_f + \xi \Sigma_{TR}},$$

and

$$R = \frac{S}{1-S} \frac{\xi \Sigma_{TR}}{\Sigma_R},$$

where Σ_R is a removal cross section. The index S makes possible the correlation of all required cross sections in the form $\sigma = \sigma_1 S^g$, except for threshold fission which may be expressed as $\sigma = \sigma_1 R^g$. A rapidly converging iterative procedure is presented through which S and R can be determined for any core composition of practical interest.

Microscopic cross section data are correlated in the above form for 43 materials using the 26-group ABBN set as parent data. The one-group method is tested for 45 different fast reactor core compositions by comparing the results of the 1-group calculation to those of 26-group fundamental-mode calculation. The results were found to agree within 1.77% in the material buckling (0.69% in k).

One-group relationships are developed for the calculation of prompt-neutron lifetime, Doppler reactivity, sodium void reactivity, and cross sections of breeder-blanket materials.

AN IMPROVED FOIL ACTIVATION METHOD
FOR DETERMINATION
OF FAST NEUTRON SPECTRA

by

Neil Alan Passman

Submitted to the Department of Nuclear Engineering
on January 22, 1971 in partial fulfillment of the requirements
for the Degree of Master of Science.

ABSTRACT

A multiple foil activation method was employed to measure the neutron spectrum in the Transistor Irradiation Facility and the Blanket Test Facility, both located at the MIT Reactor. Six of the seven foils used in this study were contained as a powdered homogeneous mixture in cylindrical capsules composed of the seventh material. The activities of all foils in each capsule were counted simultaneously using a Ge(Li) crystal and a multichannel analyzer. A PDP/8L computer and the Nuclear Data Physics Analyzer Program were used to extract the individual foil activities from the gamma ray spectrum. These activities were then fed into the SAND II code to unravel the neutron energy spectrum. The spectrum obtained for the transistor facility was compared to the one measured by the Edgewood Arsenal. The spectrum obtained for the blanket facility was compared to one computed by a 26-group ANISN calculation.

Thesis Supervisor: M. J. Driscoll

Title: Professor of Nuclear Engineering

FEASIBILITY STUDY OF A CONVERTOR ASSEMBLY FOR FUSION BLANKET EXPERIMENTS

by

Aniket Pant

Submitted to the Department of Nuclear Engineering on January 22, 1971
in partial requirement for the Degree of Master of Science.

ABSTRACT

The feasibility of using the thermal neutron flux from the MITR to obtain neutron flux spectra for fusion blanket studies was examined by considering the possible simulation of two such spectra: the uncollided 14 Mev 'source' spectrum expected from fusion machines; and the multiply - collided and backscattered fusion reactor leakage spectrum.

Two methods were investigated for producing the 14 Mev 'source' spectrum. An experimental investigation of the $\text{Li}^6 + n \rightarrow \text{T} + \text{He}$ conversion reaction showed that the first flight component of the reactor flux above 10 Mev was so much larger than the expected conversion efficiency that accentuation of the 14 Mev component of the spectrum could not be observed. A numerical investigation of a fission plate filtered by isotopically pure Li^6 absorber showed that the neutron scattering properties of Li^6 tended to increase the low energy component of the fission plate spectrum rather than decrease it. It was concluded that the simulation of a 'source' spectrum was not possible in the existing MITR facilities. Simulation of the multiply-collided leakage spectrum proved to be more practical.

The existing convertor assembly used at the MITR for LMFBR experiments was further optimized to generate a simulated fusion reactor leakage spectrum which could be used to drive a fusion blanket machine. The optimized design consisted of a 5cm thick layer of graphite followed by a 17.5cm thick UO_2 fuel region and a 0.5cm thick boral filter. Good agreement was obtained between convertor - generated and 'reference' fusion blanket spectra except in the energy range above 10 Mev where the reactor flux was too low. The calculations showed that it is feasible to conduct investigations of tritium breeding and neutron moderation and blanket heating below 10 Mev by using a modified version of the convertor assembly which is now in use for fast reactor blanket experiments.

Thesis Supervisor: M.J. Driscoll

Title: Associate Professor of Nuclear Engineering

Thesis Supervisor: L.M. Lidsky

Title: Associate Professor of Nuclear Engineering

DETERMINATION OF NEUTRON SPECTRA BY PROMPT GAMMA-RAY SPECTROMETRY

by

Charles Winfield Forsberg

Submitted to the Department of Nuclear Engineering on June 9, 1971 in partial fulfillment of the requirements for the degree of Master of Science.

Abstract

The objective of this research was the evaluation of a new type of neutron spectrometer based upon measurement and analysis of the prompt gamma-ray spectrum emitted following neutron absorption in an appropriate target. Operation of this spectrometer is based on the variation of prompt capture gamma-ray yields with incident neutron energy.

In the theoretical area, a matrix inversion method was developed to find the neutron spectrum given the measured prompt-neutron capture-gamma-ray spectrum of a target material, the target material's neutron absorption cross section as a function of energy, the dimensions of the target and the variation of the intensities of selected gamma-rays emitted by the target as a function of the incident neutron energy. Numerical tests were carried out to demonstrate the validity of the unfolding techniques.

In the experimental area, several sets of apparatus were built to extract the prompt gamma spectra of the chosen target material, tantalum-181, from a fast reactor blanket mock-up. The designed-for Ta-181 photopeak signal intensities were eventually achieved; however, high background degraded the statistical precision of these measurements sufficiently to prevent attainment of accurate final results. Information pertinent to the achievement of further improvements in equipment design was developed.

It is concluded that a neutron spectrometer based upon this principle is feasible given foreseeable improvements in experimental technique.

Thesis Supervisors: Michael J. Driscoll
Associate Professor of Nuclear Engineering

Norman C. Rasmussen
Professor of Nuclear Engineering

OPTIMIZATION OF MATERIAL DISTRIBUTIONS IN FAST BREEDER REACTORS

by

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

ABSTRACT

An iterative optimization method based on linearization and on Linear Programming is developed. The method can be used for the determination of the material distributions in a fast reactor of fixed power output, constrained power density and constrained material volume fractions that maximize or minimize integral reactor parameters which are linear functions of the neutron flux and the material volume fractions.

The method has been applied:

- (1) To the problems of optimization of the fuel distribution in the reactor core so as to obtain: (a) a maximum initial breeding gain; (b) a minimum critical mass; and (c) a minimum sodium void reactivity. Numerical results show that the same fuel distribution yields maximum breeding gain, minimum critical mass, minimum sodium void reactivity and uniform power density.
- (2) To the problem of optimization of a moderator distribution in the blanket so as to maximize the initial breeding gain. Results indicate that breeding gain is a weak function of the moderator distribution. These results are confirmed by studying the effects on the breeding gain of the insertion of a moderator, homogeneously distributed, in the blanket.

Finally, the effects on the breeding gain of surrounding the blanket by a reflector are investigated. The results show that: (a) savings in blanket thickness may be achieved with choice of a proper reflector without substantial loss in breeding gain; and (b) the transport and absorption properties of a medium, rather than its moderating properties, determine the figure of merit of a fast reactor blanket reflector.

COO-3060-2

MITNE-130

November, 1971

USE OF GAMMA SPECTROSCOPY FOR NEUTRONIC ANALYSIS OF LMFBR BLANKETS

by

Chang-Sun Kang, Norman C. Rasmussen and
Michael J. Driscoll

ABSTRACT

It was the purpose of the present investigation to extend and apply Ge(Li) gamma-ray spectroscopy to the study of fast reactor blankets. The focal point for this research was the Blanket Test Facility at the MITR and Blanket No. 2, a realistic mockup of the blanket-reflector region of a large liquid metal cooled fast breeder reactor.

It was found that Ge(Li) detectors can be simultaneously used as both high energy neutron spectrometers and continuous gamma-ray spectrometers. The broadened internal conversion spectral line at 691.4 KeV has been analyzed for the former purpose, and the Compton recoil continuum has been analyzed and unfolded for the latter. This development makes the Ge(Li) spectrometer an extremely valuable shield analysis tool.

The moisture content of the sodium chromate used in the blanket mockup has been confirmed to be less than 0.1 w/o by prompt activation analysis.

Prompt capture and inelastic gamma, and decay gamma spectra emitted by the blanket were also analyzed to perform a neutron balance with mixed results. The inability to resolve U-238 prompt capture gammas made it necessary to use the low energy Np-239 decay gammas, with the attendant uncertainties due to large self-shielding corrections. Lack of data on the variation of prompt gamma yield with neutron energy for all blanket constituents also contributed to the uncertainties, which together made it impossible to develop this method to the point where reliable practical application can be recommended.

January, 1972

NEUTRONICS OF AN LMFBR BLANKET MOCK-UP

by

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

ABSTRACT

Experimental measurements using foil activation techniques were performed on an LMFBR blanket mock-up, made up of subassemblies of 0.25-inch-diameter uranium metal fuel rods, clad in low-carbon steel, and surrounded by anhydrous sodium chromate (Na_2CrO_4) powder. This work was carried out in the MIT Blanket Test Facility which employs a converter assembly to transform the highly thermalized neutrons from the reactor's thermal column into a spectrum typical of LMFBR core leakage neutrons. Vertical and horizontal traverses were made in the blanket mock-up to confirm that an energy-independent buckling characterized the transverse leakage.

Axial reaction rate traverses of numerous materials including $\text{U}^{238}(\text{n}, \gamma)$, $\text{U}^{238}(\text{n}, \text{f})$, $\text{U}^{235}(\text{n}, \text{f})$ and $\text{Pu}^{239}(\text{n}, \text{f})$ were made throughout the blanket and reflector regions. Experimental results were compared with 26-group, S_8 calculations. In general, the agreement was good except in the iron reflector. The degree to which the U^{238} capture cross section was self-shielded was found to be the factor having the greatest affect on the calculations.

Multiple-foil packets containing resonance and threshold absorbers were irradiated at various locations within the blanket and neutron spectra unfolded from measured activities using a method based on reactor slowing-down theory. The unfolded blanket spectra were found to be harder than calculated using shielded U^{238} cross sections, but softer than calculated using unshielded U^{238} cross sections.

A STUDY OF HIGH-ALBEDO REFLECTORS FOR LMFBR'S

by

Gilbert J. Brown

Submitted to the Department of Nuclear Engineering on March 10, 1972 in partial fulfillment of the requirements for the degrees of Master of Science and Nuclear Engineer.

Abstract

High-albedo reflectors surrounding the radial blanket region of an LMFBR have been studied in conjunction with the A.E.C. supported Blanket Research Project at M.I.T. Two distinct topics were considered: (1) assessment of thermal-hydraulic effects upon overall 1000 MWe LMFBR system economics, and (2) definition of a practical configuration suitable for simulation and experimental investigation as a Blanket Mockup in the M.I.T. Blanket Test Facility.

Blanket overcooling due to the nonuniform radial power distribution in the blanket was found to have a significant economic penalty. A simple model of the blanket overcooling effect was developed, based on the degradation of the mixed mean outlet temperature and the resulting lowering of the plant thermal efficiency. It was shown that a maximum annual savings of approximately \$450,000 can be realized if the radial blanket power profile could be completely flattened, and that substantially smaller costs are associated with the beginning-to-end-of-life power shift, and excess pumping power expenditure.

These results confirm and reinforce previous neutronic-economic work at M.I.T. which showed that the use of a thin two-row blanket and a high-albedo reflector (e.g. graphite) in place of a three-row blanket and a more conventional reflector material (e.g. steel) is economically desirable, since this configuration also decreases the blanket radial power peaking. In order to permit confirmation of the calculations upon which these conclusions are based, graphite-reflected Blanket Mockup No. 3 was designed and constructed. It consists of two rows (12 inches) of blanket subassemblies reflected by two rows of graphite followed by three rows of steel. Multi-group calculations of the expected material activation traverses are compiled for comparison with the experimental data to be measured subsequently, as part of the M.I.T. Blanket Research Program.

Thesis Supervisor: Michael J. Driscoll

Title: Associate Professor of Nuclear Engineering

COO-3060-3

MITNE-129

May, 1972

INSTRUMENTAL METHODS FOR NEUTRON SPECTROSCOPY IN THE MIT BLANKET TEST FACILITY

by

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

ABSTRACT

The energy spectrum of the neutron flux in a realistic mockup of the blanket region of a large liquid-metal-cooled fast breeder reactor was measured using three different spectrometers: He-3 and Li-6 semiconductor detectors and a Proton-Recoil proportional counter. The He-3 detector was operated in the sum and difference modes, and the Li-6 detector in the sum, difference and triton modes. The experimental data was unfolded using direct, integral and derivative techniques.

Methods were developed or perfected to enable use of the He-3 detector over the neutron energy range from 10 keV to 1.3 MeV and the Li-6 detector from 10 keV to 3.1 MeV; the Proton-Recoil detector was operated in the region from 2 keV to 1.5 MeV. In general, good agreement was found between the experimental measurements for all detector types, modes of operation and methods of unfolding, except for the low-energy He-3 data.

The present experimental results and previously reported data obtained using a method based on gamma line-broadening are in relatively good agreement in the high energy region above 0.8 MeV. The measured neutron spectrum is also similar in shape to neutron spectra measured at ANL in critical assembly mockups of large LMFBR cores, but systematically softer, as expected. However, there is a large discrepancy in the energy region from 10 keV to 50 keV between the present results and either spectra unfolded from foil data or those numerically calculated using the 1-D ANISN code in the S₈ option with 26 energy groups.

ACTIVATION PROFILES IN REACTOR FUEL ELEMENTS

by

ALAN MARTIN THOMPSON

Submitted to the Department of Physics on June 2, 1972 in partial fulfillment of the requirements for the degree of Bachelor of Science.

ABSTRACT

A transparent rod model was used to develop an approximate description of the neutron flux or material activation profile within a cylindrical fuel rod in a nuclear reactor. The predicted universal intra-rod flux shape is given by the complete elliptic function of the second kind, such that:

$$\phi(r) = C_0 + C_1 \mathcal{E}\left[\left(\frac{r}{a}\right)^2\right].$$

Experiments were conducted at the MIT Reactor in the Blanket Test Facility, which accurately simulates the ambient flux spectrum and physical conditions found in a fast reactor's breeding blanket. A circular disk foil of U-238, punched into six concentric annular pieces, was placed between cylindrical UO₂ fuel pellets within a fuel rod and irradiated in the BTF. Measurement of the Np-239 and U-238 fission product activity provided data for comparison with the theoretical predictions.

Least-squares analysis of the ring data was made to compare the observed specific activity of the rings with values predicted using the elliptic function. Results indicate a good fit to the observed activity: to within an average error of approximately $\pm 2\%$.

Excellent results were obtained when this model was applied to thermal neutron irradiations of fuel rods carried out by other laboratories. An even simpler empirical relation in which flux varied as the radius to the 8/3 power was tested and found to describe intra-rod activity with substantially the same accuracy as the elliptic integral shape.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

DETERMINATION OF THE NEUTRON SPECTRUM IN THE
MITR TRANSISTOR IRRADIATION FACILITY

by

DHUNJISHAW LAL

Submitted to the MIT Chemical Engineering Department in June 1972, in partial fulfillment of the requirements for the Bachelor of Science degree.

ABSTRACT

Experiments were conducted with a He^3 neutron spectrometer to determine the neutron spectrum in the Transistor Irradiation Facility at the MIT Research Reactor. The data was analyzed by two independent methods and the results were compared with the theoretical Californium-252 fission neutron spectrum. Good agreement with the fission spectrum was observed only above about 1.75 MeV.

Further experimentation is required to resolve the discrepancies between the present and previous work in the energy range below 1 MeV.

Thesis Supervisors: Michael J. Driscoll

Title: Associate Professor of Nuclear Engineering

Johnson E. Vivian

Title: Professor of Chemical Engineering

COO-3060-4
MITNE-123
November, 1972

THE ECONOMICS OF FUEL DEPLETION IN FAST BREEDER REACTOR BLANKETS

by

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

ABSTRACT

A fast breeder reactor fuel depletion-economics model was developed and applied to a number of 1000 MWe LMFBR case studies, involving radial blanket-radial reflector design, radial blanket fuel management, and sensitivity of energy costs to changes in the economic environment.

Choice of fuel cost accounting philosophy, e.g. whether or not to tax plutonium revenue, was found to have significant effect on absolute values of energy costs, without, however, distorting design rankings, comparative results, and irradiation time optimization.

A single multigroup physics computation, to obtain the flux shape and local spectra for depletion calculations, was found to be sufficient for preliminary design and sensitivity studies. The major source of error in blanket depletion results was found to be the assumption of a fixed flux shape over an irradiation cycle; spectrum hardening in the radial blanket with irradiation is of minor importance.

The simple depletion-economics model was applied to several 1000 MWe LMFBR case studies. Advantages of a moderating reflector were found to increase as blanket thickness was reduced. For a 45 cm radial blanket, a beryllium metal reflector offered little improvement, in blanket fuel economics, over sodium; for a 15 cm blanket, beryllium increased net blanket revenue by about 60%. An improvement of about 30% in net blanket revenue resulted when each radial blanket annular region was assumed to be exposed to its own local optimum irradiation time. Optimum radial blanket irradiation time and the net blanket revenue (mills/kWh) at this optimum were found to be approximately linear in the unit fuel cycle costs, i.e. fabrication and reprocessing costs (\$/kgHM) and fissile market value (\$/kg).

HETEROGENEOUS EFFECTS IN FAST BREEDER REACTORS

by

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

ABSTRACT

Heterogeneous effects in fast breeder reactors are examined through development of simple but accurate models for the calculation of a posteriori corrections to a volume-averaged homogeneous representation. Three distinct heterogeneous effects are considered: spatial coarse-group flux distribution within the unit cell, anisotropic diffusion, and resonance self-shielding. An escape/transmission probability theory is developed which yields region-averaged fluxes, used to flux-weight cross sections. Fluxes calculated by the model are in substantial agreement with S_8 discrete ordinate calculations. The method of Benoist, as applied to tight lattices, is adopted to generate anisotropic diffusion coefficients in pin geometry. The resulting perturbations from a volume-averaged homogeneous representation are interpreted in terms of reactivities calculated from first order perturbation theory and an equivalent "total differential of k " method. Resonance self-shielding is treated by the f -factor approach, based on correlations developed to give the self-shielding factors as functions of one-group constants.

Various reference designs are analyzed for heterogeneous effects. For a demonstration LMFBR design, the whole-core sodium void reactivity change is calculated to be 90¢ less positive for the heterogeneous core representation compared to a homogeneous core, due primarily to the effects of anisotropic diffusion. Parametric studies show at least a doubling of this negative reactivity contribution is attainable for judicious choices of enrichment, lattice pitch and lattice geometry (particularly the open hexagonal lattice). The fuel dispersal accident is analyzed and a positive reactivity contribution due to heterogeneity is identified.

The results of intra-rod U-238 activation measurements in the Blanket Test Facility are analyzed and compared to calculations. This activation depression is found to be of the order of 10% (surface-to-average) for a typical LMFBR blanket rod and is ascribed to the effect of heterogeneous resonance self-shielding of U-238. Heterogeneous effects on the breeding ratio are studied with the conclusions that accounting for resonance self-shielding reduces the total breeding ratio by over 10%, but heterogeneous effects are not important for breeding ratio calculations.

SELECTION OF FOIL MATERIALS FOR
LMFBR NEUTRON SPECTROMETRY

by

Simon Yeung-Nam Ho

Submitted to the Department of Nuclear Engineering
on May 23, 1973 in partial fulfillment of the requirements
for the degree of Master of Science in Nuclear Engineering.

Abstract

The various chemical elements were assessed for use as foil detector materials for determination of LMFBR neutron spectra, with particular emphasis on those suitable for use in mixed-powder foil capsules, and where emphasis is to be placed on the sub-Kev-region, where instrumental methods are least effective.

Detailed criteria were applied to select candidate materials, 44 of which were then subjected individually to irradiation screening tests, and NaI detector counting, followed by a simulated mixed-powder application and Ge(Li) detector counting. Materials selected as having the best combination of nuclear properties were: gold, arsenic, praseodymium, manganese, lanthanum, indium, sodium, molybdenum, titanium, and rubidium.

Niobium and vanadium were identified as the preferred materials for the capsule itself.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering.

July, 1973

ASSESSMENT OF THORIUM BLANKETS FOR FAST BREEDER REACTORS

by

P. J. Wood and M. J. Driscoll

ABSTRACT

An assessment of the neutronic, economic, and engineering aspects of the use of thorium in the radial and axial blankets of Liquid Metal Cooled Fast Breeder Reactors (LMFBR's) has been performed. While the breeding performance of a thorium blanketed system has been shown to be slightly inferior to that of a comparable uranium blanketed system in terms of fissile production rate, its economic performance is significantly superior, as evidenced by up to a 30% reduction in fuel cycle costs for a 1000 MW_e reactor arising from the added 1.6 million dollars in annual income. This superiority, which arises from the projected high value of the product U-233 relative to fissile Pu in thermal spectrum reactors, is expected to persist for approximately twenty years after LMFBR commercialization, and can, therefore, significantly enhance the incentive for rapid acceptance and introduction of the breeder into utility power systems.

Detailed cash flow analysis was shown to reduce to particularly simple expressions for the fuel cycle contribution to the total cost of power:

$$\text{U-238 Blanketed System} \quad C = 0.02173 P^{49} + 0.6203$$

$$\text{Th-232 Blanketed System} \quad C = 0.07613 P^{49} - 0.04793 P^{23} + 0.6648$$

where

C = the fuel cycle contribution to the cost of power, mills/kw-hr,

P^{49} = the price of fissile Pu, \$/g, and

P^{23} = the price of U-233, \$/g.

In addition, a universal economic parameter was developed to characterize blanket performance under a wide range of economic environments. Both of the above developments will greatly reduce the amount of work required for future economic studies.

Experimental studies with thorium and uranium foils irradiated in the M.I.T. Blanket Test Facility, Blanket Mockup No. 4 were

performed to check the cross section data used in the remainder of the study, and to assign an uncertainty band to the fissile material production rate predictions.

Evaluation of the relevant physics parameters has shown that the reactor kinetics and dynamics characteristics (e.g., neutron lifetime, delayed neutron fraction, and Doppler reactivity coefficients) of the two systems are the same within calculational uncertainties, while the thorium blanketed system requires approximately 4% more core fissile material and 9% more control poison than a corresponding uranium blanketed system. Radial power gradients and blanket assembly average temporal power variations are somewhat greater for a thorium radial blanket. In-out shuffle blanket management has been shown to be the most desirable of those studies both from the economic (by a small margin) and from the thermal-hydraulic points of view.

It is concluded that LMFBR systems can be designed to accommodate uranium and thorium blankets on an interchangeable basis, and that the thorium blanket deserves strong consideration as the reference design concept for LMFBR systems.

Fast Neutron Spectrometry in
an LMFBR Blanket Reflector

by

Tsi-Pin Choong

Submitted to the department of Nuclear Engineering on
Aug. 13, 1973, in partial fulfillment of the requirements
for the degree of Master of Science.

Abstract

Previously observed anomalies involving greater than
calculated fast neutron penetration in the reflector region
surrounding mockups of fast reactor blanket were
investigated.

A Li-6 semiconductor neutron spectrometer was used to
measure neutron spectra in the range 0.1-1.0 Mev as a
function of depth into the carbon-steel reflector for depths
varying from 6 to 15 inches. The results did not confirm the
anomalous results suggested by previous threshold foil
detector traverses.

Parametric studies and diagnostic experiments indicated
that the discrepancies between calculated and measured U-238
fission traverses can be explained as the combined result of
a number of effects: errors in iron cross sections due to
the choice of weighting spectra used in their preparation,
errors in the U-238 fission cross section near threshold,
subthreshold fission in U-238, and the small amount (18ppm)
of U-235 in the depleted Uranium foils used.

Thesis Supervisor : Michael Driscoll

Title : Associate Professor of Nuclear
Engineering.

ABSTRACT

PROTON RECOIL NEUTRON SPECTROMETRY
IN A FAST REACTOR BLANKET

by

ROBERT JOHN KENNERLEY

Submitted to the Department of Nuclear Engineering on
August 13, 1973 in partial fulfillment of the
requirements for the degree of Master of Science.

Procedures and equipment for the measurement
of fast neutron spectra in fast reactor blanket
mockups using proton-recoil proportional counters were
developed, applied and evaluated. A need for improve-
ments was identified in several areas, primarily in
the detectors themselves, and in the method used to
determine their energy calibration.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

COO-2250-5

MITNE-157

January, 1974

EVALUATION OF THE PARFAIT BLANKET CONCEPT FOR FAST BREEDER REACTORS

by

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

ABSTRACT

An evaluation of the neutronic, thermal-hydraulic, mechanical and economic characteristics of fast breeder reactor configurations containing an internal blanket has been performed. This design, called the parfait blanket concept, employs a layer of axial blanket fuel pellets at the core midplane in the fuel pins of the inner enrichment zone; otherwise, the design is the same as that of the conventional LMFBR's to which the parfait configuration was compared.

Two significant advantages were identified for the parfait blanket concept relative to the conventional design. First, the parfait configuration has a 25% smaller peak fast flux which reduces wrapper tube dilation by 37% and fuel element elongation by 29%; and second, axial and radial flux flattening contribute to a 7.6% reduction in the peak fuel burnup. Both characteristics significantly diminish the problems of fuel and metal swelling.

Other advantages identified for a typical parfait design include: a 25% reduction in the burnup reactivity swing, which reduces control rod requirements; a 7% greater overpower operating margin; an increased breeding ratio, which offsets the disadvantage of a higher critical mass; and more favorable sodium voiding characteristics which counteract the disadvantage of an 8% smaller power Doppler coefficient. All other characteristics investigated were found to differ insignificantly or slightly favor the parfait design.

A FOIL-METHOD FOR NEUTRON SPECTROMETRY
IN FAST REACTORS

by

Joseph Kwok-Kwong Chan

Submitted to the Department of Nuclear Engineering on
January 15, 1974 in partial fulfillment of the requirements
for the degree of Master of Science in Nuclear Engineering.

ABSTRACT

A mixed-powder multiple foil activation method has been used to infer the neutron spectrum in a simulated fast breeder reactor blanket. After extensive screening tests, six materials were chosen as foil detectors : Au197, Mn55, Na23, As75, La139 and Pr141. A thermal calibration method was used to reduce the effect of most experimental errors. A high resolution Ge(Li) detector was used to determine the composite gamma spectrum, and the GAMANL code was used to analyze the spectrum for the individual photopeaks. Thermal-normalized activities were input to a modification of the SPECTRA program to obtain the neutron spectrum. The results were found to be in good agreement with multi-group calculations.

EVALUATION OF HIGH PERFORMANCE LMFBR BLANKET CONFIGURATIONS

by

G. J. Brown and M. J. Driscoll

ABSTRACT

The economic performance of conventional and advanced radial blanket configurations for LMFBR power reactors has been analyzed considering both fuel cycle costs and the cost penalties associated with blanket overcooling. Simple models were developed to correlate total blanket heating rates and to relate mixed-mean coolant outlet temperatures to the cost of electric power. It was found, empirically, that the total blanket heating rate could be split into two terms: one proportional to local fission rate, and the other (independent of local fission rate) representing exponential attenuation of gamma and neutron leakage from the core. The economics of thermal-hydraulic design were quantified by relating changes in power-peaking factors to changes in mixed-mean coolant temperature, which were in turn related to cycle thermal efficiency and pumping power requirements, which determined the change in net electric power available for sale, and hence the incremental change in the cost of power.

A variety of blanket-reflector configurations were examined: one, two and three subassembly rows; depleted, natural and low-enrichment uranium fuel; steel and graphite reflectors; region-wide and row-by-row orificing. An individually orificed two-row depleted-uranium-fueled, graphite-reflected blanket was found to yield the largest savings ($\$1.4 \times 10^6/\text{yr}$) relative to the base-case three-row depleted-uranium-fueled steel-reflected, region-orificed blanket.

An experimental verification of the neutronic analysis of a graphite-reflected blanket was performed using the Blanket Test Facility at the M.I.T. Research Reactor. The results of this effort indicated that the nuclear design of graphite-reflected blankets can be accomplished at least as well as that of conventional steel-reflected blankets.

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MITNE-164
August, 1974

GAMMA HEATING MEASUREMENTS IN FAST BREEDER REACTOR BLANKETS

by

P. A. Scheinert and M. J. Driscoll

ABSTRACT

Gamma heating measurements have been performed in a mockup of the blanket and reflector regions of an LMFBR using thermoluminescent dosimeters (TLD's). Supporting work was carried out on the use of cavity ionization theory to develop the spectral response factors necessary for the interpretation of the data.

Dose traverses were made using ^7LiF TLD rods encapsulated in stainless steel (to represent fuel rod cladding), aluminum (to simulate sodium coolant) and lead (to simulate UO_2 fuel). Absolute dose rates were determined using a Co-60 calibration facility developed for the purpose, and the results were compared to state-of-the-art calculations using the ANISN computer program in the S_8 , P_1 option and a 40 group (22 neutron, 18 gamma) coupled cross section set. Coolant and clad heating rates were underpredicted by roughly 50%, but the much larger fuel dose rates were predicted within the experimental uncertainty ($\pm 1\sigma = 8\%$), so that the overall gamma heating rate is only underestimated by about 20%.

Traverses made using stainless steel ionization chamber dosimeters confirm the TLD data within experimental uncertainty. It is concluded that TLD methods; with only slight and foreseeable improvements, are satisfactory for gamma heating studies in fast breeder reactor assemblies.

THE APPLICATION OF PERTURBATION THEORY AND
VARIATIONAL PRINCIPLES TO FAST REACTOR FUEL MANAGEMENT

by

Robert Edward Masterson

Submitted to the Department of Nuclear Engineering on
August 12, 1974 in partial fulfillment of the requirements
for the degree of Master of Science in Nuclear Engineering.

ABSTRACT.

Three types of perturbation methods: first order, generalized and variational, are compared to the conventional depletion time step method for calculation of system reactivity and the core and blanket breeding ratios of fast breeder reactors during burnup. A new method is developed for generating time dependent nuclide concentrations for applications in which region-averaged fluxes change linearly as a function of irradiation time. Difference equations are developed which permit simple implementation of this method, and which reduce to the conventional representation when the flux does not change over the time step.

It is shown by both numerical examples and analytic means that, in the range of interest for demonstration and commercial fast breeder reactors, most burnup related parameters change very nearly linearly as a function of time at full power: fertile and fissile nuclide concentrations, system reactivity and the components thereof, system and region-wise breeding ratios, and the region-wise averaged total neutron fluxes.

The variationally-based form of perturbation theory, as developed by Stacey, coupled with the present linear flux-change depletion model was found to predict burnup reactivity changes and core and blanket breeding ratios within $\pm 1\%$ of the conventional method. This combination is, therefore, quite useful for simple, reliable and inexpensive fast reactor fuel management computations. First order and generalized perturbation theory are shown to be less accurate, but still useful for selected applications.

Thesis Supervisor: Michael J. Driscoll
Associate Professor of Nuclear Engineering

A STACKED-FOIL METHOD FOR EPITHERMAL
NEUTRON SPECTROMETRY

by

M. K. YEUNG

Submitted to the MIT Department of Nuclear Engineering in December 1974 in partial fulfillment of the requirements for the degree of Master of Science.

ABSTRACT

A multi-foil-stack method has been developed for determination of the shape of the epithermal (1 ev to 5 kev) flux in fast reactor assemblies. A demonstration application was performed using gold foils in a fast reactor blanket mockup.

A computer program was developed to calculate multigroup self-shielding factors for stacked foils, including the effect of Doppler Broadening, given ENDF tabulated resonance parameters as input. An unfolding technique was also developed to infer the neutron spectrum from the measured induced gamma activities of the foils in the stack.

Spectra unfolded from foil data were found to be in reasonably good agreement with the calculated blanket spectrum and also with the spectrum in a 1/E calibration facility.

Thesis Supervisor: Michael J. Driscoll

Title: Associate Professor of Nuclear Engineering

COO-2250-13

MITNE-168

February, 1975

THE EFFECT OF REACTOR SIZE ON THE BREEDING ECONOMICS OF LMFBR BLANKETS

by

A. Tagishi and M. J. Driscoll

ABSTRACT

The effect of reactor size on the neutronic and economic performance of LMFBR blankets driven by radially-power-flattened cores has been investigated using both simple models and state-of-the-art computer methods. Reactor power ratings in the range 250 to 3000 MW_e were considered. Correlations for economic breakeven and optimum irradiation times and blanket thicknesses have been developed for batch-irradiated blankets.

It is shown that at a given distance from the core-blanket interface the fissile buildup rate per unit volume remains very nearly constant in the radial blanket as (radially-power-flattened, constant-height) core size increases. As a consequence, annual revenue per blanket assembly, and breakeven and optimum irradiation times and optimum blanket dimensions, are the same for all reactor sizes.

It is also shown that the peripheral core fissile enrichment, hence neutron leakage spectra, of the (radially-power-flattened, constant-height) cores remains essentially constant as core size increases. Coupled with the preceding observations, this insures that radial blanket breeding performance in demonstration-size LMFBR units will be a good measure of that in much larger commercial LMFBR's.

PARFAIT BLANKET CONFIGURATIONS FOR FAST BREEDER REACTORS

by

Richard Alward Pinnock

Submitted to the Department of Nuclear Engineering on May 9, 1975 in partial fulfillment of the requirements for the degree of Master of Science.

Abstract

An evaluation has been performed of the neutronic, thermal-hydraulic, and mechanical effects of design variations in a liquid metal cooled fast breeder reactor (LMFBR) containing internal blankets of the so-called parfait configuration in which a thin, disk-shaped region of blanket fuel replaces core fuel in the inner enrichment zone of a conventional LMFBR.

Increasing the thickness of the axial blanket while simultaneously decreasing the active core height reduced the fissile plutonium inventory, out increased the peak fast flux, peak power density, and reactivity swing. The breeding ratio was unaffected. Moving the inner blanket 5 cm. off-center toward the coolant outlet caused power generation to increase in the coolant inlet half of the core and decrease in the coolant outlet half of the core. This produced the desired reduction in peak clad temperature in the inner core zone but was accompanied by a slight increase in fissile plutonium inventory and peak fast flux and an 11.5% increase in peak power density. Reactivity swing increased slightly and breeding ratio remained essentially constant. It is concluded that these two design alternatives are not attractive, primarily because of the increase in peak power density.

Favorable results were obtained by removing the stainless steel hexagonal wrappers from the inner core subassemblies and increasing the fuel volume fraction from 30% to 35%, and then to 40%. These changes were accompanied by a small reduction in fissile plutonium inventory and significant reductions in local peak burnup. The 35 volume percent fuel design showed a 44% increase in mid-cycle breeding gain and a 51% drop in reactivity swing. The 40 volume percent fuel design showed a 67% increase in breeding gain and a 73% drop in reactivity swing, compared to the normal parfait. Breeding ratios as high as 1.40 were obtained, compared to a value of 1.20 calculated for a conventional core.

Simple models were developed to compare thermal bowing and irradiation-induced bowing of the normal parfait core to that in a conventional LMFBR core, both with freestanding fuel assemblies. It was found that the parfait design would experience maximum thermal and irradiation-induced bowing in the inner core only 42 and 16%, respectively, of that in the conventional core; corresponding figures in the outer core region were 80 and 103%.

It was found that the reduced bowing and the flat axially-integrated power generation profiles of the parfait configuration would allow both closer assembly spacing and elimination of the hexagonal subassembly ducts:

a combination which would, in turn, allow the increased fuel volume fractions discussed above. The COBRA III C thermal-hydraulics program combined with a simple model was used to evaluate bypass flow in the open lattice design. The inner core having 40 volume percent fuel was found to suffer a 6.6% reduction in the mixed mean coolant temperature rise across the core - larger than in a PWR but probably tolerable. It is concluded that the open lattice, high-fuel-volume parfait-blanket LMFBR is an attractive alternative to conventional LMFBR designs, particularly since it is not possible (without incurring compensatory penalties) in the latter to reduce bowing or flatten power to the extent that a similar design option would be practicable.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

THE BREEDING-ECONOMIC PERFORMANCE OF FAST REACTOR BLANKETS

by

Mahmoud Ketabi

Submitted to the Department of Nuclear Engineering, May 12, 1975, in partial fulfillment of the requirements for the degree of Master of Science in Nuclear Engineering.

ABSTRACT

Key fast reactor blanket fuel management parameters (breakeven and optimum irradiation times and maximum revenue) are correlated as a function of variables characterizing the economic environment under the assumption of constant local fissile buildup rate. Two cost accounting philosophies are examined: in Method A post-irradiation transactions are not capitalized (revenue from the sale of plutonium is taxed as ordinary income; reprocessing is treated as a tax deductible expense in the year it occurs); in Method B, post-irradiation transactions are capitalized.

It is shown that only two variables can be used to correlate all results: the discount rate and a lumped parameter which combines all seven of the other variables. Simple correlations are derived and fit to cash-flow-method computer program calculations, and shown to reproduce the exact results within several percent for a set of cases which cover a wide range of all independent variables involved.

Methods A and B lead to significantly different results, which can have a profound effect on breeder reactor blanket design and fuel management decisions. Method A gives rise to shorter breakeven and larger (by around 40 %) optimum irradiation times and higher revenue than Method B. Hence Method B would lead to thinner blanket designs and more rapid refueling. Sensitivity studies involving all independent variables show that Method B results are also more sensitive to changes in the economic environment, particularly the income tax rate. In the limit of public utility financing (zero tax rate), Methods A and B are identical.

The fuel management strategy employed (batch, region scatter, out-in, in-out) is shown to have no effect on the results, given an invariant local fissile buildup rate as assumed in this work.

Supervisor: Michael J. Driscoll
Associate Professor of Nuclear Engineering

IMPROVED RADIOPHOTOLUMINESCENCE
TECHNIQUES FOR GAMMA HEATING
DOSIMETRY

by

RICHARD A. MORNEAU

Submitted to the Department of Nuclear Engineering on
August 5, 1975 in partial fulfillment of the requirements
for the degree of Master of Science.

Abstract

A simple and inexpensive device for measurement of gamma dose based on the radiophotoluminescence (RPL) response of LiF thermoluminescent dosimeters (TLD's) has been constructed and evaluated. Important features of the unit include the use of photon counting methods to increase sensitivity and precision and careful selection of filters for the excitation and emission spectra.

Based upon the proven capabilities of the present design, and improvements suggested by tests carried out on the prototype unit, it is concluded that the LiF/RPL approach can be a useful supplement to, or substitute for, TLD methods: in particular for the measurement of gamma heating rates in fast reactor media, the principal application motivating the present work.

Since the RPL method is non-destructive, normal TLD readout of the LiF dosimeters can be employed following RPL readout, an important advantage.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

DESIGN OF A GRAPHITE REFLECTOR ASSEMBLY
FOR AN LMFBR

by

CRAIG A. CHAMBERS

Submitted to the Department of Nuclear Engineering
on August 11, 1975, in partial
fulfillment of the requirements for the degree of
Master of Science in Nuclear Engineering

ABSTRACT

Mechanical, neutronic/photonic, thermal/hydraulic, and economic analyses are performed to evaluate the use of graphite as a radial reflector material in the LMFBR.

A reference assembly configuration is developed, consisting of graphite discs stacked in a cylindrical stainless steel can which is in turn inserted into a standard stainless steel assembly duct. The assembly is cooled by sodium flowing upward between the duct wall and the can which clads the graphite cylinder. The limiting factor identified in the design is the questionable ability of the graphite to retain structural integrity under the high fast neutron fluences which will be experienced. Currently available test data limits the assured lifetime of even the newer high-strength isotropic graphites to a maximum of 5 years. Rotation of the assembly at midlife would be necessary to achieve even this modest lifetime. Thermal and hydraulic limits on the assembly, on the other hand, are easily accommodated by the reference design. In particular, radiant heat transfer provides inherent protection against overheating the graphite.

The promise of graphite reflectors as an economically superior alternative to additional blanket assemblies or to stainless steel reflector assemblies is not borne out. Analysis indicates that because of the relatively short nvt lifetime of graphite, the material and fabrication costs offset the slightly improved plutonium yield of the radial blanket attributable to the use of a graphite reflector. Although economic comparisons are subject to prevailing market conditions, the general conclusion can be reached that an additional blanket row is economically superior to both graphite and stainless steel reflectors: while it is still unprofitable, the net cost is less. Graphite does, however, hold a slight economic advantage over stainless steel, and the feasibility of employing graphite in current LMFBR

designs is entirely within the capability of state-of-the-art materials and procedures. Experimental verification of higher nvt tolerance would significantly improve its prospects for LMFBR service.

Thesis Supervisor: Michael J. Driscoll

Title: Associate Professor of Nuclear Engineering

BREEDING-ECONOMICS OF FBR BLANKETS HAVING
NONLINEAR FISSILE BUILDUP HISTORIES

by

DOUGLAS BRUYER

Submitted to the Department of Nuclear Engineering,
August 26, 1975, in partial fulfillment of the requirements
for the degree of Master of Science in Nuclear Engineering.

ABSTRACT

Physics-depletion-economics calculations have been carried out on several different fast reactor blanket configurations to determine and compare several important properties of the blankets, such as breakeven and optimum irradiation time and enrichment.

Particular attention is paid to the effects of realistic non-linear fissile buildup histories on the above fuel-management-related characteristics. Previous analytic methods applicable to cases in which fissile buildup is a linear function of irradiation time are shown to incur appreciable error under certain defined circumstances. A simple correction is derived to correct for this deficiency, and demonstrated to be satisfactory using illustrative numerical examples.

Supervisor: Michael J. Driscoll
Associate Professor of Nuclear Engineering

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February, 1976

GAMMA HEATING IN LMFBR MEDIA

by

M. S. Kalra and M. J. Driscoll

ABSTRACT

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

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

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

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MITNE-178
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URANIUM SELF-SHIELDING IN FAST REACTOR BLANKETS

by

O. K. Kadiroglu and M. J. Driscoll

ABSTRACT

The effects of heterogeneity on resonance self-shielding are examined with particular emphasis on the blanket region of the fast breeder reactor and on its dominant reaction - capture in U-238. The results, however, apply equally well to scattering resonances, to other isotopes (fertile, fissile and structural species) and to other environments, so long as the underlying assumptions of narrow resonance theory apply.

The heterogeneous resonance integral is first cast into a modified homogeneous form involving the ratio of coolant-to-fuel fluxes. A generalized correlation (useful in its own right in many other applications) is developed for this ratio, using both integral transport and collision probability theory to infer the form of correlation, and then relying upon Monte Carlo calculations to establish absolute values of the correlation coefficients.

It is shown that a simple linear prescription can be developed for the flux ratio as a function of only fuel optical thickness and the fraction of the slowing-down source generated by the coolant. This in turn permitted derivation of a new equivalence theorem relating the heterogeneous self-shielding factor to the homogeneous self shielding factor at a modified value of the background scattering cross section per absorber nucleus. A simple version of this relation is developed and used to show that heterogeneity has a negligible effect on the calculated blanket breeding ratio in fast reactors.

PARFAIT BLANKET SYSTEMS EMPLOYING MIXED PROGENY FUELS

by

David Charles Aldrich

Submitted to the Department of Nuclear Engineering on May 7, 1976
in partial fulfillment of the requirements
for the degree of Master of Science

ABSTRACT

A series of breeding performance comparisons has been performed for conventional vs. parfait (i.e., internally-blanketed) LMFBR systems employing mixed progeny fuels. Seven reactor configurations were investigated: Conventional cores with uranium (or thorium) external blankets, Parfait cores with uranium (or thorium) internal and external blankets, a Parfait core with uranium internal and thorium external blankets, a thorium-blanketed Parfait core having BeO in its internal blanket, and a Pu/Th-fueled conventional core with thorium external blankets.

While the parfait systems did not exhibit any direct breeding ratio advantage per se, they do provide significant performance advantages that could be exploited to achieve improved breeding capability. The relative performance advantages of the uranium-blanketed parfait cores over their conventional counterparts was similar to findings in previous work at MIT; the relative advantages of the thorium parfait cores over their conventional counterparts appear even more favorable [reduced peak fast flux (28%), burnup reactivity swing (22%), peak power density (7%), radial fast flux and power gradients in inner core zone (60%)].

Addition of BeO to the inner blanket region led to degraded neutronic performance and is not considered a favorable design alternative.

Strategies for improving breeding and/or doubling time performance are discussed. Analysis based on heat removal capability suggests the following relationships:

$$\frac{\Delta T}{T} \approx \frac{-\Delta p}{p} - \frac{2}{g} \frac{\Delta \eta}{\eta}$$

where

$$\frac{\Delta \eta}{\eta} \approx 0.04 \frac{\Delta \epsilon}{\epsilon}$$

and

$$\frac{\Delta p}{p} \leq R \frac{1}{\left(1 + \frac{\Delta \epsilon}{\epsilon}\right)} - 1$$

T = doubling time, p = specific power, g = breeding gain, ϵ = average enrichment, R = peak to average power in conventional core. Rough calculations indicate maximum doubling time reductions of up to 28% (considerably less than the 50% claimed in recent French studies).

Fissile inventory and production rate data has been compiled for use in future comparative economic analyses. The thorium-blanketed parfait system and the Pu/Th-fueled system are of interest because of their relatively large U-233 production rates.

EXPERIMENTAL VERIFICATION OF
BREEDING PERFORMANCE OF
FAST REACTOR BLANKETS

by
Shin-Shyong Wu

Submitted to the Department of Nuclear Engineering, Massachusetts Institute of Technology, in June 1976, in partial fulfillment of the requirements for the degrees of Master of Science in Nuclear Engineering and Nuclear Engineer.

ABSTRACT

Fast reactor blanket mockup experiments are analyzed with particular attention to assessment of breeding performance. The spectral index $\sigma_C^{28}/\sigma_f^{25}$ (or the related parameter C_b^*) is shown to be a particularly useful parameter for characterization of breeding-related properties.

The measured value of $\sigma_C^{28}/\sigma_f^{25}$ in a mockup of a typical oxide-fueled LMFBR blanket was found to be 0.102 ± 0.005 , which is slightly lower than the value of 0.110 calculated using state-of-the-art multigroup methods.

Parametric studies of the effect of various changes on the calculated value of $\sigma_C^{28}/\sigma_f^{25}$ were carried out. Errors attributable to the effects of heterogeneity or blanket composition were discredited as likely sources of the C/E discrepancy : experimental technique and the capture cross section of U-238 were identified as the most probable reasons.

The results are consistent with other recent findings that calculated LMFBR breeding ratios are slightly higher than those verified by experimental determinations; but more definitive experiments and calculations are still called for.

Thesis Supervisor : Michael J. Driscoll
Title : Associate Professor of Nuclear Engineering

COO-2250-25
MITNE-199
March, 1977

EVALUATION OF ADVANCED FAST REACTOR BLANKET DESIGNS

by

J. I. Shin and M. J. Driscoll

ABSTRACT

Various fast reactor blanket design concepts - moderated, fissile seeded, alternatively fueled, etc. - have been evaluated to develop a more comprehensive understanding of the technical basis for improved breeding and economic performance. Simple analytical models and equations have been developed, verified by state-of-the-art computer calculations, and applied to facilitate interpretation and correlation of blanket characteristics.

All design concepts examined in the study could be fit into a self-consistent methodology: for example, the fissile buildup histories of all blanket compositions and regions, from a single pin to an entire blanket, could be fitted to the same dimensionless correlation, and all blankets have the same dimensionless optimum irradiation time.

It was also found that the external breeding ratio at the beginning of blanket life is a constant multiple of the external breeding ratio averaged over life for an optimally irradiated blanket. Hence one can use beginning-of-life studies to correctly rank the breeding performance of various blanket design options.

Although oxide fuel was found to have economically favorable characteristics given equal fabrication costs per unit mass of heavy metal, thin (2-row) UC-fueled blanket concepts appear to be slightly preferable under projected short-term future economic conditions due to their excellent neutronic and thermal-hydraulic characteristics, and their narrow margin of deficiency even under conditions favoring oxide fuel.

It is confirmed that the non-linear fissile buildup history characteristic of FBR blankets must be considered in making sufficiently accurate fuel management decisions for real reactors, and it is shown that a batch fuel management option produces about 15% less plutonium than other commonly considered options such as In-Out shuffling.

All results were consistent with the observation that very little improvement in external blanket breeding performance can be envisioned unless core design changes are allowed. Breeding ratio improvements were often detrimental to blanket economics (and vice versa).

RESONANCE REGION NEUTRONICS OF UNIT CELLS IN FAST AND THERMAL REACTORS

by

A. A. Salehi, M. J. Driscoll and O. L. Deutsch

ABSTRACT

A method has been developed for generating resonance-self-shielded cross sections based upon an improved equivalence theorem, which appears to allow extension of the self-shielding-factor (Bondarenko f-factor) method, now mainly applied to fast reactors, to thermal reactors as well.

The method is based on the use of simple prescriptions for the ratio of coolant-to-fuel region-averaged fluxes, in the equations defining cell averaged cross sections. Linearization of the dependence of these functions on absorber optical thickness is found to be a necessary and sufficient condition for the existence of an equivalence theorem. Results are given for cylindrical, spherical and slab geometries. The functional form of the flux ratio relations is developed from theoretical considerations, but some of the parameters are adjusted to force-fit numerical results. Good agreement over the entire range of fuel and coolant optical thicknesses is demonstrated with numerical results calculated using the ANISN program in the SgP₁ option. Wider application of these prescriptions, to fast and thermal group applications, is suggested.

The present results are shown to include the Dancoff approximation and Levine factor results, developed previously for thermal reactors, as limiting cases. The theoretical desirability of correcting for the effects of neutron moderation in the fuel region of fast reactor unit cells is demonstrated: a refinement not required in thermal reactors.

The method is applied to U-238 self-shielding in thermal and fast reactor applications. Heterogeneity corrections in fast reactors are so small that the method is not severely tested. Calculations of PWR unit cells are compared with LEOPARD program calculations. Epithermal group cross sections for U-238 calculated from the LIB-IV fast reactor cross section set using the present method agree with the LEOPARD results within about $\pm 1\%$ for typical PWR lattices; and while disagreement is larger for larger and smaller unit cells, all of the qualitative features of group cross section dependence are in agreement.

FAST REACTOR BLANKET BENCHMARK CALCULATIONS

by

MARTIN STANLEY MACHER

Submitted to the Department of Nuclear Engineering
on September 26, 1977 in partial fulfillment of
the requirements for the Degree of
Master of Science in Nuclear Engineering.

ABSTRACT

Fast reactor blanket mockups irradiated to date in the MIT Blanket Test Facility are described. Geometric configurations and dimensions are presented together with region-wise homogenized compositions in a form suitable for use in benchmark computer calculations.

Data sets and computer programs selected and put into use to update the state-of-the-art LMFBR computational capabilities at MIT are described, as are two utility codes written by the author to facilitate operation of the overall system.

Illustrative calculations using the new data sets are compared with older results and with measured fast neutron penetration (threshold foil detector traverses) in the reflector of Mockup No. 5B. Calculated integral parameters differ by as much as 15%, and the measured fast neutron flux continues to be much larger than calculated at deep penetrations. Suggestions relative to future efforts to reduce the discrepancy are made, centering on the calculational methodology.

Thesis Supervisor: Michael J. Driscoll

Title: Associate Professor of Nuclear Engineering

GENERATION OF FEW GROUP CROSS SECTIONS FOR THERMAL REACTORS
USING FAST REACTOR METHODS

by

Habib Aminfar

Submitted to the Department of Nuclear Engineering on May 11, 1978 in partial fulfillment of the requirements for the degrees of Master of Science in Nuclear Engineering and Nuclear Engineer.

ABSTRACT

Epithermal (>0.6 ev) cross sections prepared using thermal reactor (LEOPARD) and fast reactor (ANISN) preparation codes are compared for PWR lattices as a function of fuel-to-moderator ratio. The fast reactor approach is based on the shielding factor (f factor) method and a new equivalence theorem relating the background cross section per shielded nucleus, σ_b , in heterogeneous and homogeneous unit cells.

Systematic differences of 5% to 10% in fissile and fertile absorption cross sections are found above 5 Kev, and discrepancies as large as 30% in fissile cross sections are evident between 0.6 ev and 5 Kev. Although sensitivity studies of the effect on the overall multiplication factor indicate that the differences are self compensating to a considerable degree, reasons are developed for preferring the fast reactor methodology.

It is concluded that the fast reactor method can be adapted to serve both the thermal and fast applications given a modest amount of additional work.

Thesis Supervisor:
Title:

Michael J. Driscoll
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EXPERIMENTAL COMPARISON OF THORIUM AND URANIUM
BREEDING POTENTIAL IN A FAST REACTOR BLANKET

by
David William Medeiros

Submitted to the Department of Nuclear Engineering on May 12, 1978, in partial fulfillment of the requirements for the degree of Bachelor of Science.

ABSTRACT

An experimental comparison of thorium and uranium breeding potential in a simulated fast reactor blanket has been performed. A (partially) thorium-fueled subassembly was constructed, and was irradiated in the uranium-fueled Blanket Test Facility of the M.I.T. Research Reactor. Data obtained from uranium and thorium foils irradiated in the subassembly traversing fuel rod and in the adjacent thermal spectrum facility were used in measuring the ratio of the integral (spectrum-averaged) microscopic cross-section of Th-232 to that of U-238 for both fission and capture (with primary emphasis on the latter) in a neutron spectrum typical of the radial blanket spectrum of a fast breeder reactor. State-of-the-art computer calculations of the same parameters were also executed (ANISN: S_0P_1 using the fifty-group LIB-IV cross section set). Results were as follows:

$$\left(\frac{\bar{\sigma}_c}{\sigma_c}\right)_{28}^{\text{EXPT}} = 1.331 \pm 0.021; \quad \left(\frac{\bar{\sigma}_c}{\sigma_c}\right)_{28}^{\text{CALC}} = 1.183$$

$$\left(\frac{\bar{\sigma}_f}{\sigma_f}\right)_{28}^{\text{EXPT}} = 0.346 \pm 0.046; \quad \left(\frac{\bar{\sigma}_f}{\sigma_f}\right)_{28}^{\text{CALC}} = 0.217$$

Reasons for the discrepancies are discussed, and suggestions are made for improvements in both the experiments and the calculations; the calculations and their cross-section input are most in need of re-evaluation with respect to capture ratio calculations, while the experimental fission ratio data needs improvements in both precision and data analysis.

These results indicate that thorium may be approximately thirty percent more efficient an absorber than U-238; hence, utilization of thorium as a fertile material may permit more neutronically efficient breeding blanket designs.

Thesis Supervisor: Michael J. Driscoll
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DESIGN OF A THORIUM METAL
FILLED BLANKET ASSEMBLY

by

FRANK REMSEN FIELD, III

Submitted to the Department of Nuclear Engineering
on May 12, 1978 in partial fulfillment of the requirements
for the Degree of Bachelor of Science.

ABSTRACT

The thermal-hydraulic design characteristics of an LMBFR internal blanket assembly fueled with a thorium-uranium metal alloy are evaluated and compared with standard UO_2 -fueled assemblies.

Th-U fuel pins, with or without sodium bonding, operate well within steady state materials limits, and can be directly substituted for UO_2 pins of the same diameter. They remain much cooler during loss of flow accidents accompanied by scram: under conditions where UO_2 -fueled assemblies approach sodium boiling, the Th-U assembly has a 600°F margin to boiling. On the other hand, the Th-U pins heat up twice as fast during extremely rapid overpower transients, or loss of flow without scram.

The results are sufficiently encouraging to warrant further consideration of Th-U fueled internal blanket assemblies in LMFBRs, where they can provide beneficial spectrum-hardening, serve as a heat sink during certain transients, and provide for self-denaturing of the bred fissile species.

THESIS SUPERVISOR: Michael J. Driscoll

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INTERFACIAL EFFECTS IN FAST REACTORS

by

M. S. Saidi and M. J. Driscoll

ABSTRACT

The problem of increased resonance capture rates near zone interfaces in fast reactor media has been examined both theoretically and experimentally.

An interface traversing assembly was designed, constructed and employed to measure U-238 capture rates near the blanket-reflector interface in the MIT Blanket Test Facility. Prior MIT experiments on a thorium-uranium interface in a blanket assembly were also reanalyzed. Extremely localized fertile capture rate increases of on the order of 50% were measured immediately at the interfaces relative to extrapolation of asymptotic interior traverses, and relative to state-of-the-art (LIB-IV, SPHINX, ANISN/2DB) calculations which employ infinite-medium self-shielding throughout a given zone.

A method was developed to compute a spatially varying background scattering cross section per absorber nucleus, σ_0 , which takes into account both homogeneous and heterogeneous effects on the interface flux transient. This permitted use of the standard self-shielding factor method (Bondarenko f-factors) to generate modified cross sections for thin layers near the interfaces. Calculations of the MIT experiments using this approach yielded good agreement with the measured data.

AN EVALUATION OF THE BREED/BURN FAST REACTOR CONCEPT

by

B. Atefi, M. J. Driscoll and D. D. Lanning

ABSTRACT

A core design concept and fuel management strategy, designated "breed/burn", has been evaluated for heterogeneous fast breeder reactors. In this concept internal blanket assemblies after fissile material is bred in over several in-core cycles, are shuffled into a moderated radial blanket and/or central island. The most promising materials combination identified used thorium in the internal blankets (due to the superior performance of epithermal Th-U233 systems) and zirconium hydride (ZrH_{16}) as the moderator (because of the compact assembly and core designs it permitted).

The advantage of moving the U233-enriched thorium internal blankets to the zirconium-hydride-moderated radial blanket included production of 20-30% of the total system power by the radial blanket, which resulted in a 10-15% reduction in the peak core linear heat generation rate (LHGR). This in turn can be translated into either a 10-15% increase in the total power production from the core or a 15-20% reduction in the core fuel assembly fabrication requirements for the same amount of energy delivered. Other advantages of this core include a 40% reduction in the total reprocessing requirements, and a 25% reduction in the transportation and reprocessing of the plutonium-bearing assemblies compared to a more conventional heterogeneous core on the U-Pu cycle with no blanket shuffling or moderation.

The reduced plutonium handling and the ability to denature U-233 with U-238 offer a potential improvement in proliferation resistance.

The alternative shuffling strategy of moving the enriched internal blankets to the middle of the core, and creation of a near-critical moderated central island resulted in a 27% reduction in the total core fissile plutonium requirement, a 50% reduction in the total reprocessing requirements, and a 60% reduction in the transportation and reprocessing of plutonium-bearing assemblies.

It was found that the unit price of plutonium has a large effect on the levelized fuel cycle cost. At a plutonium price of 27 \$/gr (the indifference value of plutonium in LWRs) the core with the central island had a 20% lower fuel cycle cost compared to the reference heterogeneous core. This reduction in fuel cycle cost increased to 40% when plutonium was assigned zero value.

It is concluded that the breed/burn fuel management concept is a useful addition to the FBR core designer's repertoire of variations which can be worked into the same basic core frame.

THE NEUTRONICS OF
INTERNALLY HETEROGENEOUS
FAST REACTOR ASSEMBLIES

by

MARTIN GERARD PLYS

Submitted to the Department of Nuclear Engineering
on May 8, 1981 in partial fulfillment of
the requirements for the Degrees of Bachelor of Science
and Master of Science in Nuclear Engineering

ABSTRACT

The objective of this research has been to determine the relative advantages of adding internal moderation and/or blankets to fast reactor assemblies. This study of small-scale, intra-assembly heterogeneity was motivated by the success of large-scale, inter-assembly heterogeneity in recent fast reactor and fast-mixed spectrum reactor research. The primary benefit of the large-scale heterogeneity is a reduction in sodium void worth.

This work was limited to modelling of single-assembly behavior, as opposed to whole core studies. A well-benchmarked driver assembly uniformly fueled with mixed oxide pins was chosen as the base case. Selected fuel pins were replaced by various combinations of moderator and blanket pins of the same size, the sole exception being a design employing oversized blanket pins. The special pins were arranged in two generic fashions: A "Ringed Assembly" design which symmetrically distributed these pins within the assembly and a "Flux Trap" design which clustered these pins at the assembly center.

Absolute results were found to be sensitive to the assembly representation (composition of mesh triangles) and choice of axial buckling, while relative results were by and large unaffected. Therefore the relative performance of each assembly design could be confidently ranked.

It is concluded that only the so-called Ringed Moderator design, which is moderated but unblanketed, shows any clear advantages with respect to the base case. In general, it was found that any gains in the reduction of sodium void-worth are linked with heavy metal loading penalties.

Thesis Supervisor: Dr. Michael J. Driscoll

Title: Professor of Nuclear Engineering

THERMAL-HYDRAULIC DESIGN OF
INTERNALLY HETEROGENEOUS
FAST BREEDER REACTOR ASSEMBLIES

by

SAGHIR AHMAD

Submitted to the Department of Nuclear Engineering, July 1981
in partial fulfillment of the requirements for the Degree
of Nuclear Engineer and the Degree of Master of Science
in Nuclear Engineering.

ABSTRACT

The objective of this investigation has been to evaluate the thermal-hydraulic characteristics of internally heterogeneous Liquid-Metal Cooled Fast Breeder Reactor (LMFBR) assemblies. The state-of-the-art code, COBRA-IV-I, has been employed for this purpose. This study was done on isolated single assemblies as opposed to whole core analysis. A well-benchmarked driver assembly uniformly fueled with mixed oxide pins was chosen as the base case against which all variations were compared.

Four different configurations (including the uniformly fueled assembly) have been analyzed. The three heterogeneous assemblies were: centrally blanketed (depleted uranium replacing mixed oxide inside fuel pins); centrally moderated and blanketed (depleted uranium pins surrounding a cluster of zirconium hydride pins); and a composite fuel assembly (containing large-diameter blanket pins). The results obtained under similar performance conditions - same system pressure, coolant inlet temperature, coolant mass flow rate, and average heat flux for the assembly - indicate that the uniform fuel assembly and the centrally moderated and blanketed assembly function within the design limits on coolant, clad and fuel centerline temperature, but the centrally blanketed and the composite fuel assemblies will have to be derated to compensate for driver fuel displacement. The centrally moderated and blanketed assembly appears attractive because of its more favorable radial temperature distribution and its enhanced safety characteristics, e.g., the inventory of cold sodium in the center of the assembly should be beneficial under overpower conditions. However, this also has unfavorable aspects (thermal

stress, shock and striping) because of the large sodium temperature differential within the assembly. Isolated subchannel analysis was also investigated and found to be too conservative to be used to make practical design decisions.

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