27
8/28/78
25 to NTIS

MASTER

MASTER

Preliminary Neutronic Study of Actinide Transmutation in a Fast Reactor

M. L. Williams
J. W. McAdoo
G. F. Flanagan

MASTER

OAK RIDGE NATIONAL LABORATORY
OPERATED BY UNION CARBIDE CORPORATION • FOR THE DEPARTMENT OF ENERGY

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy \$4.50 ; Microfiche \$3.00

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, contractors, subcontractors, or their employees, makes any warranty, express or implied, nor assumes any legal liability or responsibility for any third party's use or the results of such use of any information, apparatus, product or process disclosed in this report, nor represents that its use by such third party would not infringe privately owned rights.

Contract W-7405-eng-26

Engineering Physics Division

PRELIMINARY NEUTRONIC STUDY OF ACTINIDE
TRANSMUTATION IN A FAST REACTOR

M. L. Williams
J. W. McAdoo*
G. F. Flanagan

*UCC-ND Computer Sciences Division

Date Published - August 1978

NOTICE This document contains information of a preliminary nature.
It is subject to revision or correction and therefore does not represent a
final report.

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

104

ABSTRACT

Preliminary results are cited from a neutronic study of actinide recycle in a fast reactor, using an LMFBR model. The work was performed as part of a Chemical Technology Division program for assessing the merit of actinide transmutation as a means of waste disposal. Calculations were performed by coupling two-dimensional diffusion theory calculations with point-depletion recycle calculations. Cross sections were obtained from a "preliminary" version of ENDF/B-V.

Results are given concerning the recycled-actinide equilibrium condition, and the waste-actinide destruction rate. The effect of adding U-238 diluent to reduce power peaking is examined parametrically. The heterogeneous effects of placing an actinide target assembly in a fuel lattice are also examined.

I. INTRODUCTION

The Chemical Technology Division at ORNL is currently involved in a large-scale program to assess the incentives for actinide partitioning and transmutation by fission as a means of reducing the long-term risks represented by nuclear wastes.¹ The effort involves a broad spectrum of disciplines, of which neutronic analysis is a relatively small, though vital, component. The field of neutronic analysis has been further subdivided into a study of transmutation in thermal reactors and in fast reactors (LMFBR's). The two programs at present are essentially independent, with the former being pursued at Savannah River Laboratory and the latter at Oak Ridge National Laboratory.

The present report documents the initial phase of the ORNL fast reactor study. The goals of the program are to (1) determine the effective actinide fission transmutation rate, (2) evaluate the effect of actinide recycle on reactor performance, (3) examine factors which influence the transmutation efficiency, and (4) provide any necessary analytical support in neutronics to Chem Tech.

The calculations which are quoted are predominately scoping studies meant to identify problems and reveal any technical difficulties that may be encountered in their resolution. A major difficulty in such a preliminary analysis is that the results depend strongly on the assumed recycle scenerio, which of course is not presently fixed. Iteration between the neutronic studies and the partitioning studies must occur. The conclusions in this report -- while tentative -- represent the zeroth iteration, and refinements to the basic assumptions will be treated as perturbations to these base case results.

II. CROSS-SECTION PREPARATION

One of the first problems encountered in any LMFBF neutronics study involving trans-plutonium isotopes is the lack of accurate nuclear data. Most of the previous studies (see reference 2 for a review of the numerous investigations) have used data from ENDF/B-III,³ ENDF/B-IV,⁴ ENDL,⁵ or from data compilations by Benjamin,⁶ or others.⁷

The majority of heavy actinide cross sections in these sets were obtained by "backing out" integral information from chemical analysis of thermal-reactor spent fuel. Knowing the quantity of a given actinide present and the reactor flux, evaluators can assign some effective cross section to the appropriate parent nuclides. Because samples irradiated in a fast reactor spectrum are quite scarce, the accuracy of actinide cross sections in the aforementioned sets tends to fall off in the higher energy range. Much of the available high-energy data is from the analysis of products produced in nuclear explosions.

The next version of ENDF/B to be released -- version V -- supposedly contains substantial improvement in the heavy-actinide cross sections over the fast-reactor energy range. For this reason, it was decided in this study to use an unofficial, "pre-preliminary" set of ENDF/B-V actinide data that was then available at ORNL for data testing.⁸ During the cross-section processing, the following discrepancies in the preliminary data were found:

- a) the ^{241}Am cross section was inaccurate,
- b) the (n, 2n) cross sections for some nuclides were missing, and
- c) discontinuities existed at the interface of the resolved and unresolved resonance ranges.

These problems were indicated to the ENDF/B-V Evaluation Committee and will be corrected before the official Version V data is released. For the present study, the difficulty with the ^{241}Am cross section was overcome by merely replacing the ENDF-V value with a value from ENDF-IV. The other two problems were judged to introduce errors that were within the tolerance of this preliminary study. A more detailed study should either examine the effect of these errors or use the official ENDF-V data, which should be available shortly. The nuclides for which data were taken from the preliminary ENDF-V set are listed in Table 1.

Table 1. Nuclides Processed into Cross-Section Libraries
(from pre-preliminary ENDF/B-V data)

232U	236Np	236Pu	240Am	241Cm	249Bk
233U	237Np	237Pu	241Am	242Cm	250Bk
234U	238Np	238Pu	242Am	243Cm	
235U	239Np	239Pu	243Am	244Cm	249Cf
236U	236Np M	240Pu	244Am	245Cm	250Cf
237U		241Pu	242Am M	246Cm	251Cf
238U		242Pu	244Am M	247Cm	252Cf
239U		243Pu		248Cm	253Cf
		244Pu		249Cm	
					253Es

Point data from the preliminary ENDF-V tape were processed to multigroup format using the MINX⁹ processing code. A 171-group, infinitely dilute library was first generated, and then -- collapsing with zone-averaged fluxes -- the following zone-dependent libraries were created:

- i) 51-group ANISN¹⁰ format library
- ii) 51-group ISOTX¹¹ format library
- iii) 5-group CITATION¹² format library
- iv) 1-group ORIGEN¹³ format library

In addition, several ENDF-III and IV cross sections that had been generated previously for LMFBR studies were added to the master library. These data were for the following materials:

238U (detail resonance treatment)
 239Pu (detail resonance treatment)
 240Pu (detail resonance treatment)
 241Pu (detail resonance treatment)
 242Pu (detail resonance treatment)
 235U (detail resonance treatment)
 O
 Na
 SS-316
 B

Since most recycle calculations were performed with the ORIGEN computer code, it is interesting for data testing purposes to compare the one-group actinide cross sections in the "old" ORIGEN library (from 1968 AI cross sections averaged over a single fast spectrum) to the newer values in the preliminary ENDF/B-V ORIGEN library. Tables 2 and 3 list comparisons of the capture and fission cross sections, respectively.¹⁴

The general trend in these tables is for important nuclides (e.g., ²⁴¹Am, ²⁴³Am, ²⁴⁴Cm, ²³⁸Pu, etc.) in the new library to have higher capture-to-fission ratio than in the old ORIGEN library. Unfortunately, this is a negative trend for effective actinide transmutation, since it increases the buildup of higher actinides and decreases their burnup.

Table 2. Comparison of (n, γ) Cross Sections (barns) from Old ORIGEN Library and "Preliminary"-Preliminary ENDF/B-V Data

Nuclide	Previous ORIGEN LMFBR	New ORIGEN Value (collapsed with spectrum from indicated zone ^a)				
		Core			Axial Blanket ^b	Radial Blanket ^c
		CZ1	CZ2A	CZ2B		
²⁴¹ Am	0.990	2.27	1.80	2.03	3.83	4.08
²⁴² Am	0.403	.105	.0668	.0865	.227	.246
²⁴² Am M	0.403	.520	.395	.457	.907	.978
²⁴³ Am	0.555	1.21	.938	1.07	2.12	2.31
²⁴⁹ Bk	0.385	1.20	.794	1.00	2.77	3.15
²⁴⁹ Cf	0.831	.407	.327	.367	.676	.731
²⁵⁰ Cf	0.416	.453	.346	.401	.915	1.06
²⁵¹ Cf	0.411	.324	.273	.297	.511	.560
²⁵² Cf	0.388	.292	.249	.269	.444	.470
²⁵³ Cf	0.400	.281	.174	.229	.638	.702
²⁴¹ Cm	0.0	.222	.185	.202	.339	.359
²⁴² Cm	0.380	.498	.359	.430	.992	1.09
²⁴³ Cm	0.400	.255	.204	.229	.427	.463
²⁴⁴ Cm	0.373	.870	.733	.797	1.33	1.41
²⁴⁵ Cm	0.400	.342	.295	.317	.481	.505
²⁴⁶ Cm	0.302	.261	.207	.233	.444	.475
²⁴⁷ Cm	0.362	.305	.265	.283	.441	.473
²⁴⁸ Cm	0.306	.268	.216	.243	.511	.587
²⁵³ Es	0.0	.331	.192	.265	.942	1.12
²³⁷ Np	0.765	1.85	1.50	1.67	2.96	3.15
²³⁸ Np	0.360	.118	.0748	.0969	.255	.277
²³⁶ Pu	0.220	.616	.474	.545	1.06	1.13
²³⁷ Pu	0.0	.185	.163	.173	.231	.236
²³⁸ Pu	0.224	1.01	.799	.902	1.65	1.74
²⁴² Pu	0.342	.505	.424	.463	.790	.870
²⁴³ Pu	0.568	.404	.324	.363	.677	.729
²⁴⁴ Pu	0.340	.260	.181	.222	.591	.661
²³⁴ U	0.450	.631	.522	.576	1.03	1.12
²³⁶ U	0.663	.595	.499	.547	.952	1.02
²³⁷ U	0.410	.561	.415	.489	1.11	1.22

^aSee Fig. 2 for zone identification.

^bAveraged over ABZA and ABZB.

^cAveraged over RB1 and RB2.

III. LMFBR MODEL DEFINITION

A waste management scheme consisting of partitioning/transmutation of the actinides is viewed by most authorities as a relatively long-term solution (~20 years) to the problem of radioactive waste disposal.² An LMFBR model to be used in transmutation studies should therefore be representative of an advanced type of reactor that will be operative around the year 2000.

The reactor model for the present studies was selected to be a 1200 Mw advanced-oxide-fueled reactor, with a high specific power and a core burnup value about 15% higher than the value presently predicted

Table 3. Comparison of (n,fission) Cross Sections from Old ORIGEN Library and "Preliminary"-Preliminary ENDF/B-V Data

Nuclide	Previous ORIGEN LMFBR	New ORIGEN Value (collapsed with spectrum from indicated zone ^a)				
		Core			Axial Blanket ^b	Radial Blanket ^c
		CZ1	CZ2A	CZ2B		
²⁴¹ Am	.463	.252	.284	.277	.170	.161
²⁴² Am	1.83	.619	.395	.510	1.35	1.48
²⁴² Am M	1.83	5.14	4.41	4.77	7.38	7.80
²⁴³ Am	.237	.218	.251	.243	.131	.122
²⁴⁹ Bk	.133	.152	.174	.169	.0915	.0853
²⁴⁹ Cf	2.55	2.76	2.48	2.62	3.70	3.87
²⁵⁰ Cf	1.22	1.01	1.10	1.06	.780	.754
²⁵¹ Cf	2.03	2.27	2.16	2.21	2.71	2.84
²⁵² Cf	1.24	.709	.688	.713	.873	.920
²⁵³ Cf	2.00	.736	.466	.606	1.65	1.83
²⁴¹ Cm	0.0	3.3	3.05	3.20	4.29	4.47
²⁴² Cm	.420	.160	.184	.179	.0941	.0874
²⁴³ Cm	.320	2.66	2.43	2.55	3.48	3.67
²⁴⁴ Cm	.412	.430	.451	.439	.284	.273
²⁴⁵ Cm	2.45	2.75	2.54	2.64	3.45	3.58
²⁴⁶ Cm	.300	.254	.293	.284	.155	.145
²⁴⁷ Cm	2.15	1.85	1.87	1.86	1.84	1.86
²⁴⁸ Cm	.293	.289	.328	.319	.193	.186
²⁵³ Es	0.0	0.0	0.0	0.0	0.0	0.0
²³⁷ Np	.360	.315	.358	.346	.205	.194
²³⁸ Np	3.10	.693	.442	.571	1.51	1.67
²³⁶ Pu	1.40	1.84	1.71	1.79	2.22	2.29
²³⁷ Pu	0.0	3.51	3.40	3.45	3.68	3.69
²³⁸ Pu	1.38	1.19	1.17	1.19	1.24	1.25
²⁴² Pu	.278	.320	.363	.352	.210	.199
²⁴³ Pu	2.03	.890	.795	.848	1.26	1.34
²⁴⁴ Pu	.280	.163	.187	.181	.103	.0965
²³⁴ U	.510	.307	.347	.336	.204	.193
²³⁶ U	.116	.0964	.111	.108	.0564	.0523
²³⁷ U	1.82	.576	.596	.587	.530	.525

^aSee Fig. 2 for zone identification.

^bAveraged over AB2A and AB2B.

^cAveraged over RB1 and RB2.

for a commercial LMFBR. These assumptions favor high transmutation rates, and indeed it may be that special "high performance" reactors of this type will be developed to serve as transmuter reactors. The model geometry was based on a proposed G.E. design for a commercial LMFBR,¹⁵ with minor changes made to the material composition. Figures 1 and 2 show the hex and R-Z configurations, respectively, for this reactor, and Tables 4 and 5 give the beginning-of-life nuclide concentrations for the respective geometries.

The selected fuel management scheme is described in Table 6. Using this refueling schedule, the composition of the refuel material was determined iteratively with an enrichment search to find the minimum amount of Pu required to keep the reactor critical over all burn cycles. These calculations were performed with the "quasi-equilibrium" option in the CITATION diffusion code, using the five-group cross-section library and the RZ model previously discussed. The resulting refuel composition is given in Table 7. Refuel plutonium was assumed to have been recycled from a advanced LWR, and the uranium was assumed to be enrichment plant tails (see Table 6 for isotopic compositions).

When the refuel composition had been fixed, the equilibrium condition for the RZ model was computed with a full-blown CITATION depletion calculation. Neutron balance was maintained both by adjusting control boron density and by adjusting the k-eigenvalue. There did not appear to be much difference in the equilibrium condition based on the two methods. Since the control material adjustment actually depends on specification of a control-rod pattern, it was decided to use the results from the k-eigenvalue method. The reactor composition at the beginning of equilibrium cycle is described in Table 8; and one-group, zone-averaged fluxes computed at four time steps during an equilibrium cycle are given in Table 9.

Some reactor performance parameters based on the beginning of equilibrium composition are in Table 10. Finally, Table 11 gives some burnup parameters for the equilibrium configuration.

Table 4. Composition of Reactor Assemblies, kg

	Inner Core	Outer Core	Axial Blanket	Radial Blanket
²³⁵ U	0.14	0.13	0.10	0.41
²³⁸ U	56.93	54.40	40.08	163.97
²³⁹ Pu	6.24	7.38	0	0
²⁴⁰ Pu	2.85	3.37	0	0
²⁴¹ Pu	1.60	1.93	0	0
²⁴² Pu	0.62	0.83	0	0
SS-316	25.12	25.12	15.56	27.0
Na	5.65	5.65	3.50	6.87
φ	8.77	8.77	5.43	22.04

Table 5. Homogenized Beginning-of-Life-Nuclide Concentrations for RZ Model

Nuclide	Concentration (kg)			
	Inner Core	Outer Core	Axial Blanket	Radial Blanket
²³⁵ U	2.5356 + 01 ^a	2.4231 + 0.	1.7749 + 01	9.4983 + 01
²³⁸ U	1.0247 + 04	9.7917 + 03	7.2149 + 03	3.8370 + 04
²³⁹ Pu	1.1226 + 03	1.3277 + 03	0.0	0.0
²⁴⁰ Pu	5.1285 + 02	6.0657 + 02	0.0	0.0
²⁴¹ Pu	2.8828 + 02	3.4707 + 02	0.0	0.0
²⁴² Pu	1.2392 + 02	1.4921 + 02	0.0	0.0
Clad (SS-316)	4.5221 + 03	4.5215 + 03	2.8001 + 03	6.3152 + 03
Na	1.0163 + 03	1.0162 + 03	6.2933 + 02	1.6073 + 03
φ	1.5786 + 03	1.5784 + 03	9.7743 + 02	5.1579 + 03
Control Channel	1.3266 + 02	4.7373 + 01	3.2852 + 02	0.0
Control Rod	4.5279 + 02	1.6169 + 02	0.0	0.0

^aRead as 2.5356 x 10¹.

Table 6. Assumed Fuel Management Program for LMFBR

Core Refuel	1/3 IC and 1/3 OC every 270 full-power days
Axial Blanket Refuel	1/3 AB every 270 full-power days
Radial Blanket Refuel	1/6 RB every 270 full-power days
Refuel U composition	99.75% ²³⁸ U, 0.25% ²³⁵ U
Refuel Pu composition	55% ²³⁹ Pu, 25% ²⁴⁰ Pu, 14% ²⁴¹ Pu, 6% ²⁴² Pu
Average core burnup, at discharge	116,000 MW-d/T
Capacity factor	0.80

Fig. 1. Core Layout in a Proposed G.E. Design for a Commercial LMFBR.

CALCULATION MODEL 1200 MWe

NUMBER OF HEXAGONAL ASSEMBLIES

RADIAL REGION	1	2	3	4	5	6	7	TOTAL
ZONE 1 FUEL	18	162						180
ZONE 2 FUEL			84	96				180
RADIAL BLANKET					72	78	84	234
CONTROL	1	13	5*					19
TOTAL	19	175	89	96	72	78	84	613

*THE FIVE CONTROL ASSEMBLIES ASSIGNED TO THE INNER REGION OF CORE ZONE 2 ARE DERIVED FROM THE RATIO OF SIDES PRESENTED TO THE APPROPRIATE REGION FOR THOSE CONTROL ASSEMBLIES ON THE INTERFACE BETWEEN CORE ZONES.

Fig. 2. R-Z Configuration of a Proposed G.E. Design for a Commercial LMFBFR.

Table 7. Composition of LMFBR Refuel Material
(No Recycle)

Nuclide	Composition (kg)			
	Inner Core	Outer Core	Axial Blanket	Radial Blanket
²³⁵ U	8.0988	7.5234	1.1558 + 1 ^a	1.5831 + 1
²³⁸ U	3.2682 + 3	3.0438 + 3	4.698 + 3	6.3950 + 3
²³⁹ Pu	3.4835 + 2	4.6532 + 2		
²⁴⁰ Pu	1.5690 + 2	2.1264 + 2		
²⁴¹ Pu	8.8198 + 1	1.1942 + 2		
²⁴² Pu	3.7918 + 2	5.1262 + 1		

^aRead as 1.1558 x 10¹.

Table 8. Reactor Composition at Beginning of Equilibrium Cycle (RZ Model)

Nuclide	Composition (kg)				
	Inner Core	Outer Core	Axial Blanket #1 (Over Inner Core)	Axial Blanket #2 (Over Outer Core)	Radial Blanket
²³⁵ U	1.7315 + 1 ^a	1.7216 + 01	1.5223 + 1	1.5125 + 01	8.17512 + 01
²³⁸ U	9.329 + 03	8.781 + 03	7.0978 + 03	6.7878 + 03	3.7725 + 04
²³⁹ Pu	1.065 + 03	1.3223 + 03	9.876 + 01	7.0784 + 01	5.52608 + 02
²⁴⁰ Pu	4.8893 + 02	6.4664 + 02	3.8064	1.9506	1.68742 + 01
²⁴¹ Pu	2.0122 + 02	2.8306 + 02	9.7566 - 02	3.638 - 02	3.52889 - 01
²⁴² Pu	1.13948 + 02	1.5418 + 02	1.6908 - 03	4.5087 - 04	5.2446 - 03
Clad (SS-316)	4.52214 + 03	4.52147 + 03	2.800 + 03	2.7997 + 03	6.31515 + 03
Na	1.01634 + 03	1.0162 + 03	6.2933 + 02	6.2925 + 02	1.60734 + 03
Oxygen	1.57855 + 03	1.57835 + 03	9.7744 + 02	9.773 + 02	5.15786 + 03
Control Channel	3.656 + 02	1.30213 + 02	3.2852 + 02	1.1732 + 02	0.0
Control Rod	1.87677 + 02	6.742 + 01	0.0	0.0	0.0
Fission Products	2.1753 + 02	2.2033 + 02	7.9197	4.9275	4.145 + 01

^aRead as 1.7315 x 10¹.Table 9. Time Dependent Zone-Averaged Flux^a
(cm²·sec)⁻¹ During Equilibrium Cycle

Location	No. of days from beginning of cycle			
	0.0	90	180	270
Inner core	6.2314 + 15 ^b	6.6517 + 15	7.0135 + 15	7.3274 + 15
Outer core	5.5258 + 15	5.3171 + 15	5.1304 + 15	4.9675 + 15
Axial blanket (Inner core)	1.5914 + 15	1.7296 + 15	1.8579 + 15	1.9829 + 15
Axial blanket (Outer core)	1.2781 + 15	1.2663 + 15	1.2502 + 15	1.2424 + 15
Radial blanket	7.6880 + 14	7.3780 + 14	7.1063 + 14	6.8795 + 14

^aOne-group flux, obtained by summing over five energy-group fluxes found with CITATION.^bRead as 6.2314 x 10¹⁵.

Table 10. Reactor Performance Parameters at Beginning of Equilibrium Cycle

Power	1200 MWe
Efficiency	.35
Specific Power	153 kW/kg heavy metal in core
Maximum Power Density	783 W/cc
Average Power Density across Core	456 W/cc
Reactivity	.03703
Breeding Ratio	1.198 ^a

^aFertile material assumed to be ²³⁸U, ²⁴⁰Pu; fissile material assumed to be ²³⁵U, ²³⁹Pu, ²⁴¹Pu.

Table 11. Reactor Burnup Data

Total Power	3430 MWth
Specific Power	153 $\frac{\text{kW}}{\text{kg HM in core}}$
Average Core Burnup	116,000 MWd/T
Power produced in Inner Core	1660 MWth
Specific Power, Inner Core	148 kW/kg HM IC
Average Burnup, Inner Core	120,000 MWd/T
Power produced in Outer Core	1550 MWth
Specific Power, Outer Core	139 kW/kg HM OC
Average Burnup, Outer Core	112,000 MWd/T

IV. ACTINIDE RECYCLE SCHEME

This study examined the feasibility of recycling actinide waste from one PWR and one LMFBF. In outlining the specific program for actinide recycle, it is convenient to divide the actinide inventory into two categories, (1) Fresh and (2) Target.

"Fresh actinides" are defined as those actinides (excluding U and Pu) removed following each cycle from the fuel and blanket assemblies of the LMFBF, and from the fuel assemblies of the PWR. "Target actinides" are those which have been previously recycled in special target assemblies.

Since the fuel management program consists of refueling 1/3 of the core each cycle (also assumed for the PWR) at any given time there are three "batches" of fuel (and hence actinides) present in the reactor, differentiated by the irradiation period. Every 337.5 days (270 full-power days), a batch of fuel which has been burned for 1012.5 days (3 cycles) is removed from the reactor and sent to reprocessing where it remains for one additional cycle. The rotation of batches is illustrated in Fig. 3.

Fig. 3. Rotation of Reactor Fuel Batches.

This spent fuel will contain fission products and some amount of fresh actinides, which is partitioned from the fission products and the valuable U and Pu isotopes, and is then sent to a special actinide fabrication plant. Table 12 gives the amount of fresh actinides removed from the LMFBR and the PWR, respectively, every 337.5 days.

The composition of the actinide waste produced by the LMFBR was found to be quite dependant on assumptions regarding the model definition. There are two predominant transmutation paths for the generation of actinides heavier than plutonium:

The specific isotopic composition of the heavy actinide waste depends on the relative importance of each buildup mechanism. In particular, the ratio of the ^{243}Am to ^{241}Am depends both on the initial

Table 12. Fresh Actinide Discharge^a by Model
LMFBR and PWR

Nuclide	LMFBR		PWR	
	kg	Wt %	kg	Wt %
²³⁷ Np	3.77	12.0	14.10	66.0
²⁴¹ Am	10.86	34.6	3.27	15.3
²⁴² Am M	0.67	2.1	0	0
²⁴³ Am	12.52	39.8	2.30	10.8
²⁴² Cm	0.84	2.7	.08	.4
²⁴³ Cm	.07	.2	3.0×10^{-3}	.014
²⁴⁴ Cm	2.46	7.8	1.59	7.40
²⁴⁵ Cm	.23	.7	.02	.094
²⁴⁶ Cm	8.8×10^{-3}	.03	2.2×10^{-3}	.01
Total	31.4	100	21.4	100

^aDischarged every 270 full-power days. Charge = 1/3 core refueled, Burnup ~115,000 MWd/T (LMFBR), ~30,000 MWd/T (LWR).

percentages of ²⁴¹Pu and ²⁴²Pu in the fuel, and also the LMFBR burnup, since both ²⁴¹Pu → ²⁴²Pu → ²⁴³Pu → ²⁴³Am and ²⁴¹Am → ²⁴²Am → ²⁴³Am provide loss mechanisms for ²⁴¹Am and production terms for ²⁴³Am.

The plutonium fuel in this study had a relatively large percentage of the heavier plutonium isotopes (see Table 6), implying the use of recycled plutonium from a high-burnup LWR. The use of LMFBR fuel that is relatively rich in ²⁴¹Pu and ²⁴²Pu tends to enhance the contribution from ²⁴³Pu, especially for high burnup values when the ²⁴¹Pu is transmuted more efficiently into ²⁴²Pu. In this particular study, the LMFBR waste discharged contained 15% more ²⁴³Am than ²⁴¹Am; this contrasts to a GE study which calculated 30% less ²⁴³Am than ²⁴¹Am. Part of the difference can be traced to assumptions regarding the LMFBR burnup and the fuel composition. However, the use of the old ORIGEN value for the ²⁴¹Am cross section also accounts for some of the disagreement.

V. ACTINIDE-RECYCLE EQUILIBRIUM CONDITION

After some period of recycle, the total actinide inventory will reach an equilibrium state, which corresponds to a zero growth rate for actinide waste produced by the LWR and the LMFBF. At this point the mass of waste actinides remains constant and is distributed among the LMFBF, the LWR, and the reprocessing plant. When these reactors have reached the end of their lifetimes, the actinides contained in them must either be buried or transferred to other transmuting reactors.

There are basically three questions regarding the recycled-actinide equilibrium condition which must be addressed:

- (a) How much time is required for the recycled waste actinides to reach equilibrium?
- (b) What is the equilibrium mass and composition?
- (c) What factors can affect the equilibrium condition?

A very simple analytic approach to the recycle problem can perhaps provide insight into the determination of the equilibrium condition. Waste actinides (which of course do not include the U and Pu contained in the fuel and blanket assemblies) are produced by capture reactions in fuel assemblies (and due to the low flux, to a much lesser extent in the blanket assemblies) and are "burned" by fission reactions in the fuel and the target assemblies. The process can be simplified even further, if we consider the idealized case in which the fresh actinides produced in the fuel assemblies are continuously reprocessed and instantaneously recycled. With this assumption, actinides can only be burned in target assemblies and can only be produced in fuel assemblies. The equation for the lumped actinide waste can then be written approximately as

$$\frac{dA}{dt} = (\sigma_c^F \bar{\phi}_F)F - (\sigma_f^A \bar{\phi}_A)A, \quad (1)$$

where

- A = mass of waste actinides in target assemblies
- σ_f^A = effective actinide fission cross section
- F = mass of U and Pu in fuel assemblies

σ_c^F = effective actinide "buildup" cross section

$\bar{\phi}_A$ = average flux in actinide target assembly

$\bar{\phi}_F$ = average flux in fuel assembly.

Note that actinide decay and capture do not appear in the equation since these processes do not result in a net gain or loss in the total actinide inventory. There is no discrete actinide feed term due to the assumption of continuous reprocessing. The effective actinide buildup cross section is defined such that $\sigma_c^F \bar{\phi}_F$ = probability per unit time, per unit fuel mass, that some heavy actinide will be produced. The effective actinide fission cross section is defined such that $\sigma_f^A \bar{\phi}_A$ = probability per unit time, per unit mass of heavy actinide, that some actinide will fission.

The value for F is actually time-dependant due to depletion, but the conversion process and the refueling schedule do not allow F to vary far from some mean value. Therefore we will assume that F is constant in time. The "effective cross sections" are also time-dependant because of composition and spectrum charges over a cycle, but for this rough calculation let us assume that these also are constant. The solution to Eq. (1) is then

$$A(t) = A_0 e^{-(\bar{\phi}_A \sigma_f^A)t} + \left[1 - e^{-(\bar{\phi}_A \sigma_f^A)t} \right] \left(\frac{\sigma_c^F}{\sigma_f^A} \right) \left(\frac{\bar{\phi}_F}{\bar{\phi}_A} \right) F \quad (2)$$

The equilibrium value for the actinide inventory is

$$\left(\frac{\sigma_c^F}{\sigma_f^A} \right) \left(\frac{\bar{\phi}_F}{\bar{\phi}_A} \right) F \quad (3)$$

- Thus we see that the steady state actinide mass is proportional to
- (a) the mass of fuel (i.e., U + Pu) producing actinides,
 - (b) the ratio of effective actinide buildup to effective actinide-fission cross section,
 - (c) the target assembly self-shielding factor $\frac{\bar{\phi}_F}{\bar{\phi}_A}$

The rate at which equilibrium is approached depends only on the magnitude of $\bar{\phi}_A \sigma_f^A$. Although the preceding derivation is very simplified, it does point out some of the factors which should be examined in recycle studies.

The actual equilibrium actinide composition for this study was found using the ORIGEN point-depletion computer code. The region-averaged flux from Table 9 was input to ORIGEN, and the recycle scheme described in the previous section was applied for 13 recycles, i.e., 52 reactor-refueling cycles. After this many recycles most of the actinide isotopes have reached equilibrium, exceptions being the higher Z nuclides such as Californium. Since these nuclides represent only a small fraction of the total actinide inventory, the total actinide mass has essentially stabilized. Figure 4 shows the time-dependent actinide inventory. It is apparent that 95% equilibrium is achieved after about eight recycles.

The steady-state actinide mass is divided among the three irradiation batches in the reactor (0-, 1-, and 2-cycle exposure) and a fourth batch (3-cycle exposure) contained in the out-of-reactor fuel cycle. A flow chart depicting the overall scenario for actinide recycle has been suggested by Croff,² and is included in this report as Fig. 5. The mass composition for the fresh LMFBR fuel stream is given in Table 7; the actinide composition of the spent fuel stream is given in Table 12; and the composition of the steady-state target assembly discharge (i.e., the recycled actinides shipped to reprocessing) is given in Table 13.

The net equilibrium concentrations of all three batches of recycled actinides contained in the LMFBR are given in Table 14. The entire mass of actinides in the target assemblies at equilibrium is approximately

Fig. 4. Time-Dependent Composition of Actinide Inventory.

Fig. 5. Actinide Recycle Flow Chart.

Table 13. Steady-State Target Assembly Discharge
Sent to Reprocessing Plant

Element	kg	Weight Fraction of Total
U	1.82	$1.43 \cdot 10^{-2}$
Np	10.0	$7.86 \cdot 10^{-2}$
Pu	49.3	$3.88 \cdot 10^{-1}$
Am	31.9	$2.51 \cdot 10^{-1}$
Cm	34.2	$2.69 \cdot 10^{-1}$
Bk	$7.21 \cdot 10^{-3}$	$5.67 \cdot 10^{-5}$
Cf	$1.84 \cdot 10^{-2}$	$1.45 \cdot 10^{-4}$
Total	127.20	

^aRead as 1.43×10^{-2} .

486 kg, or about 2.2% of the total heavy metal in the core. Of the total mass of target actinides, about 13% is ^{237}Np , 32% is plutonium isotopes (mostly ^{238}Pu), 31% is Am isotopes, 23% is Cm isotopes (mostly ^{244}Cm), and the remaining 1% is shared among U, Bk, and Cf isotopes.

The actinide transmutation efficiency can be computed by comparing the amount of waste actinides remaining after recycle to the total amount produced by the LMFBR and the LWR. To this end, we define a coefficient of recycle efficiency after n reactor cycles to be

$$\eta_n = \left(1 - \frac{A_n}{nW}\right) \times 100 ,$$

where

A_n = mass of target and fresh actinides after n cycles

W = mass of fresh actinides produced per cycle by the LMFBR
and LWR

In the limit as $n \rightarrow \infty$, $A_n \rightarrow$ equilibrium value, and thus $\eta \rightarrow 100\%$, since the denominator increases linearly with n . The calculation of η for this study is summarized in Table 15. After 13 recycles, the value for A has essentially stabilized, while the "total" actinides produced is increasing linearly. The result after about 50 years of recycle is an efficiency of nearly 80%.

Table 14. Actinide Inventory in LMFBR Targets (3 Batches) after 13th Recycle (i.e., Beginning 52nd Reactor Cycle)

Nuclide	kg	Weight Fraction of Total
^{234}U	4.06	$8.36 \cdot 10^{-3}$ ^a
^{235}U	0.956	$1.97 \cdot 10^{-3}$
^{236}U	0.743	$1.53 \cdot 10^{-3}$
^{238}U	$2.73 \cdot 10^{-3}$	$5.62 \cdot 10^{-6}$
^{237}Np	62.00	$1.277 \cdot 10^{-1}$
^{239}Np	$5.27 \cdot 10^{-5}$	$1.09 \cdot 10^{-7}$
^{238}Pu	64.70	$1.332 \cdot 10^{-1}$
^{239}Pu	27.68	$5.700 \cdot 10^{-2}$
^{240}Pu	43.34	$8.925 \cdot 10^{-2}$
^{241}Pu	5.29	$1.089 \cdot 10^{-2}$
^{242}Pu	14.4	$2.966 \cdot 10^{-2}$
^{244}Pu	$9.71 \cdot 10^{-4}$	$2.00 \cdot 10^{-6}$
^{241}Am	69.48	$1.431 \cdot 10^{-1}$
$^{242}\text{Am M}$	1.72	$3.54 \cdot 10^{-3}$
^{243}Am	78.54	$1.617 \cdot 10^{-1}$
^{242}Cm	3.971	$8.18 \cdot 10^{-3}$
^{243}Cm	0.974	$2.01 \cdot 10^{-3}$
^{244}Cm	72.59	$1.495 \cdot 10^{-1}$
^{245}Cm	20.75	$4.273 \cdot 10^{-2}$
^{246}Cm	12.30	$2.533 \cdot 10^{-2}$
^{247}Cm	1.43	$2.95 \cdot 10^{-3}$
^{248}Cm	0.579	$1.19 \cdot 10^{-3}$
^{249}Bk	$1.69 \cdot 10^{-2}$	$3.48 \cdot 10^{-5}$
^{249}Cf	$4.00 \cdot 10^{-2}$	$8.24 \cdot 10^{-5}$
^{250}Cf	$1.83 \cdot 10^{-2}$	$3.77 \cdot 10^{-5}$
^{251}Cf	$2.86 \cdot 10^{-3}$	$5.89 \cdot 10^{-6}$
^{252}Cf	$2.58 \cdot 10^{-4}$	$5.32 \cdot 10^{-7}$
Total	486.61	

^aRead as 8.36×10^{-3} .

Table 15. Recycle Efficiency After 13th Recycle

kg Actinides in LMFBR	485.56
kg Actinides at Reprocessing Plant	127.20
kg Actinides Remaining After 13 Recycles	612.8
kg Actinides Produced by LWR	$1.089 \cdot 10^3$
kg Actinides Produced by LMFBR	$1.602 \cdot 10^3$
Total Actinides Produced	$2.691 \cdot 10^3$
Recycle Efficiency $\equiv (1 - \frac{612.8}{2.531 \cdot 10^3}) \times 100 = 77.2\%$	

Another parameter of interest is the equilibrium rate of actinide destruction by transmutation (fission). At steady state, a total of 180 kg of waste actinides is introduced into the LMFBR every 337-day cycle. This mass is composed of about 53 kg of fresh actinides and 127 kg of actinides previously recycled (assuming small reprocessing losses). But 127 kg is also discharged each cycle, the product of burning the 180 kg charged for 3 cycles. Therefore the average rate of actinide destruction is

$$D.R = \frac{(180 - 127) \text{ kg}}{(180) \text{ kg} (3.337 \text{ days}) \left(\frac{1 \text{ yr}}{365 \text{ days}} \right)} = 10.6\% \text{ per year of reactor residence}$$

or about 8% per calendar year.

This value is higher than the value reported by Croff² due to (a) the higher specific power of the LMFBR and (b) the greater residence time in the reactor; in other words, the higher burnup. It is interesting to note the effect of a 3-cycle refueling schedule as opposed to a 2-cycle schedule. In the former case, the actinide targets are in the reactor 75% of the time (assuming one-cycle reprocessing), while in the latter case they are exposed for only 67% of the time. For a fixed specific power, this results in a higher burnup per year for the three-cycle fuel management program. The question as to which scenario is more realistic is not presently answerable.

VI. HETEROGENEOUS EFFECTS

As was shown in Eq. (3), the equilibrium condition for recycled actinides depends on the ratio of the average flux in the actinide assemblies to the average flux in the fuel assemblies. For a homogeneous recycle scheme, this factor is obviously unity; but for the heterogeneous case, it may be either greater or less than one. If the factor is significantly greater than one, power peaking will occur in the target assembly, causing a hot spot in the core. If the factor is significantly less than one, the self-shielding of the target assembly adversely affects the rate at which actinides are burned. Spectrum shifts will also affect power production and transmutation.

Because the recycle calculations were performed with the zero-dimensional code ORIGEN, no spatial or spectral effects were observable; and therefore the previous results are essentially for the homogeneous case. It is the purpose of this section to examine the heterogeneous effects on recycle.

Calculations were performed for a cylindrical actinide cell surrounded by an infinite fuel region, using the 1-dimensional discrete ordinates code ANISN. The 51-group cross-section library was used, with a P1 Legendre expansion of the scatter cross sections. The composition of the actinide assembly was determined by replacing the heavy metal in a fuel assembly with actinide material having the equilibrium isotopic concentrations given in Table 14. An S4 quadrature set was used in the calculations.

The relative actinide flux to fuel flux was found by ratioing the mean flux in the actinide cell to the asymptotic flux in the infinite fuel region. There appeared to be little difference in the two values. This is because the neutron mean free path in a fast reactor is long enough to make the core look homogeneous. However, such is not the case for the relative power production by the target and fuel assemblies. The target assembly has a substantially higher effective fission cross section than does a fuel assembly. It was found that an actinide assembly produces about three times as much power as an average fuel assembly.

To avoid local power peaking, it is then necessary to dilute the actinide concentration in the target assembly. Most previous works have used ^{238}U as the diluent in order to minimize the perturbation to the breeding ratio. We also used ^{238}U in this preliminary study; however, it should be noted that for the assumed scenario of no plutonium being removed from the target assemblies, there is no apparent advantage to ^{238}U . In fact, ^{238}U can contribute to the buildup of waste actinides. Future study will examine the possibility of using an inert material such as aluminum or magnesium oxide.

Figure 6 shows the effect of varying ^{238}U concentration on the relative power ratio. It can be seen that 65% diluent reduces the power output of the target assembly to nearly the same value as for a fuel

Fig. 6. Relative Power vs. Percent Diluent in Target Assembly.

assembly. With a 50/50 mixture of actinides and diluent, the target assembly power is about 15% higher than for an average fuel assembly. Figure 7 shows that adding ^{238}U diluent has little effect on the relative fluxes.

Since more fissions occur in the actinide assemblies than in a fuel assembly, one might expect that the spectrum in the target assembly would be harder. Table 10 shows that this is indeed the case, and that the addition of diluent tends to soften the spectrum. At 75% diluent the fraction of flux greater than 200 keV is about the same for the fuel and target assemblies. The conclusion from the spectral analysis is that the actinide assembly -- when diluted with a sufficient amount of ^{238}U to avoid power peaking -- has approximately the same spectrum as for the homogeneous core. Therefore, spectral heterogeneous effects appear to be small for a fast reactor.

Fig. 7. Relative Flux vs. Percent Diluent in Target Assembly.

Table 16. Spectral Effect of Lumping Actinides

Case	$\frac{\phi > 200 \text{ keV}}{\phi \text{ total}}$	$\frac{\phi > 1 \text{ keV}}{\phi \text{ total}}$
Fuel	.353	.9913
100 Act/ 0 Diluent ^a	.442	.9976
90 Act/ 10 Diluent	.434	.9974
75 Act/ 25 Diluent	.422	.9971
50 Act/ 50 Diluent	.401	.9965
25 Act/ 75 Diluent	.378	.9955

^aDiluents assumed to be ²³⁸U.

If we assume that the waste actinides are diluted 65% with ^{238}U , there must be approximately 1380 kg of heavy metal to be distributed in target assemblies. This material will be placed in the inner core region where the flux is highest, and therefore about 22 inner core assembly locations must be made available for target assemblies, or about 6% of the total number of core assemblies. At equilibrium, an average inner core assembly contains about 10 kg of fuel. Thus the required fissile inventory of the LMFBR reactor is reduced by 220 kg (4%) when the actinides from itself and 1 PWR are recycled.

VII. CONCLUSIONS AND RECOMMENDATIONS

The results of this study confirm the conclusions of earlier studies that transmutation in a fast reactor is a feasible means of reducing actinide wastes. However, it was found that assumptions concerning the reactor model and fuel management program can have significant influence on specific results. In some aspects the present study was optimistic, since a high-burnup LMFBR -- the most effective transmutor reactor -- was used in the calculations; but this assumption was based on the realistic fact that reactor technology should improve considerably by the time actinide recycle is possible.

The conclusions of this preliminary study can be summarized as follows:

- (a) The trend in the ENDF/B-V "pre-preliminary" cross-section data over the standard ORIGEN library values appears to be in the negative direction as far as efficient transmutation is concerned.
- (b) For the assumed model and recycle scenario, actinide mass reaches 95% of its equilibrium value after eight recycles (~30 years)
- (c) The equilibrium actinide mass in the transmuting LMFBR is about 485 kg, which represents about 2% of the heavy metal in the core.
- (d) At steady state conditions, the actinide destruction rate is about 8% transmutation per calendar year.

(e) After approximately 50 years of recycle, the inventory of waste actinides is reduced to 80% of the value for the no-recycle case.

(f) Power peaking in target assemblies is somewhat drastic, but can be reduced to tolerable levels by the addition of about 65% ^{238}U diluent.

(g) Spectral and spatial heterogeneous effects on actinide recycle do not appear to be a great problem in an LMFBR, if an appropriate amount of diluent has been added to the target assemblies.

The results obtained thus far are certainly not totally complete; many areas in the neutronic analysis of actinide recycle in an LMFBR still require further examination. Specifically, the following recommendations are made for future study:

(a) Examine the effect on conclusions of using more recent cross section data (e.g., official ENDF/B-V).

(b) Identify quantitatively the sensitivity of final results to model and data assumptions.

(c) Examine the effect of recycle on LMFBR breeding ratio and Na void coefficient.

(d) Determine the effect of fission-product impurities in target assemblies.

(e) Study the use of other diluents, such as an inert oxide.

(f) Optimize recycle strategy relative to fuel management parameters.

(g) Iterate with the chemical reprocessing study to determine a workable recycle scenerio.

ACKNOWLEDGMENTS

The authors wish to acknowledge the contribution of M. A. Bjerke in generating the ORIGEN cross-section data, and the helpful suggestions from A. G. Croff.

REFERENCES

1. J. O. Blomeke, "A Program to Establish the Technical Feasibility and Incentives for Partitioning," Proceedings of Conference on Management of Radioactive Waste: Waste Partitioning as an Alternative, Seattle, Wash., June 1976, NR-CONF-001.
2. A. G. Croff, et al., "A Preliminary Assessment of Partitioning and Transmutation as a Radioactive Waste Management Concept," ORNL/TM-5803 (Sept. 1977).
3. D. E. Cullen and P. J. Hlavac, "ENDF/B Cross-sections," Revised Version III, BNL-17100 (Nov. 1972).
4. D. Garber (compiler), "ENDF/B Summary Documentation," RNL-17541 (ENDF-201), 2nd Edition (Oct. 1975).
5. R. J. Howerton, "The LLL Evaluated Nuclear Data Library (ENDL) Translated into ENDF/B Format," UCID-16727 (March 1975).
6. R. W. Benjamin, et al., "A Consistent Set of Transplutonium, Multi-group Cross-sections," Savannah River Laboratory, DP-MS-74-61 (March 1975).
7. E. J. Hennelly, "Nuclear Data for Actinide Recycle," DP-MS-74-62 (March 1965).
8. C. R. Weisbin, personal communication.
9. C. R. Weisbin, et al., "MINX: A Multigroup Interpretation of Nuclear X-Sections from ENDF/B," LA-6488-MS (Sept. 1976).
10. W. W. Engle, "A Users Manual for ANISN, A One-Dimensional Discrete Ordinates Code with Anisotropic Scattering," ORGDP-K-1693 (March 1967).
11. G. E. Bosler, et al., "LASIP-III, A Generalized Processor for Standard Interface Files," LA-6280-MS (April 1976).
12. T. B. Fowler and D. R. Vondy, "Computer Code CITATION," ORNL/TM-2496, Rev. 2 (July 1971).

13. M. J. Bell, "ORIGEN: The ORNL Isotope Generation and Depletion Code," ORNL-4628 (July 1973).
14. M. A. Bjerke, personal communication.
15. "Cooperative Nuclear Data and Methods Development Quarterly Report January-March 1976," GEAP-14074-3 (April 1976).

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

INTERNAL DISTRIBUTION

1. L. S. Abbott	25. C. D. Scott
2. C. W. Alexander	26. D. W. Tedder
3. R. G. Alsmiller, Jr.	27. D. B. Trauger
4. D. E. Bartine	28. J. W. Wachter
5. R. E. Blanco	29. C. R. Weisbin
6. J. O. Blomeke	30. J. R. White
7. T. J. Burns	31-35. M. L. Williams
8. W. D. Burch	36. R. G. Wymer
9-13. A. G. Croff	37. A. Zucker
14. D. E. Ferguson	38. P. Greebler (Consultant)
15. B. C. Finney	39. W. Loewenstein (Consultant)
16. G. F. Flanagan	40. R. E. Uhrig (Consultant)
17. H. Goldstein (Consultant)	41. R. Wilson (Consultant)
18. N. M. Greene	42-43. Central Research Library
19. R. E. Leuze	44. ORNL Y-12 Technical Library
20. F. C. Maienschein	Document Reference Section
21. J. W. McAdoo	45-46. Laboratory Records Department
22. R. W. Peelle	47. Laboratory Records, ORNL RC
23. S. Raman	48. ORNL Patent Office
24. RSIC	

EXTERNAL DISTRIBUTION

49. Director, Research and Technical Support Division, DOE-ORO
50. Director, Reactor Division, DOE-ORO
51. D. E. Large, DOE, Oak Ridge, Tennessee 37830
- 52-53. Director, Reactor Research and Technology Division, DOE,
Washington, D.C. 20545
54. R. D. Walton, DOE, Washington, D.C. 20545
55. G. K. Oertel, DOE, Washington, D.C. 20545
56. R. Cooperstein, Nuclear Regulatory Commission, Washington, D.C.
20555
57. A. M. Platt, Battelle Northwest, P. O. Box 999, Richland, WA 99352
58. T. C. Gorrell, E. I. duPont de Nemours and Company, Savannah
River Laboratory, Aiken, SC 29801
59. S. L. Beaman, General Electric Company, Mail Code S-34,
P. O. Box 5020, Sunnyvale, CA 94086
60. H. A. C. McKay, Atomic Energy Research Establishment, Harwell,
Didcot., Oxon., OX11 0RA, ENGLAND
61. J. R. Grover, Atomic Energy Research Establishment, Harwell,
Didcot., Oxon., OX11 0RA, ENGLAND
62. G. N. Kelly, National Radiation Protection Board, Harwell,
Didcot., Oxon., OX11 0RA, ENGLAND
63. Y. Sousselier, Centre d'etudes nucleaires de Fontenay-aux-Roses,
B. P. No. 6, Fontenay-aux-Roses, FRANCE
64. M. Takahashi, OECD-Nuclear Energy Agency, 38, Boulevard Suchet,
F-75016 Paris, FRANCE

65. H. O. Haug, Institut fur Heisse Chemie Kernforschungszentrum,
Postfach 3640, 75 Karlsruhe, FEDERAL REPUBLIC OF GERMANY
66. L. Tondinelli, Lab. Teoria e Calcolo Reottori, CNEN - CSN
Casaccia, CP 2400, I-00100 Rome, ITALY
67. F. Girardi, Commission of the European Communities, CCR Euratom,
I-21200 Ispra, ITALY
68. J. O. Liljenzin, Chalmers University of Technology, Department
of Nuclear Chemistry, S-402, 20 Goteborg, SWEDEN
- 69-95. Technical Information Center (TIC).