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BAW-3867-1
November 1967

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FAST BREEDER REACTOR
STATIC PHYSICS METHODS DEVELOPMENT
AND ANALYSIS PROJECT

— Quarterly Technical Report —

July - September 1967

Babcock & Wilcox

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BAW-3867-1
November 1967
TID-4500
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— Quarterly Technical Report —

July - September 1967

by

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Department

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AEC Contract No. AT(30-1)-3867
B&W Contract No. 859-3105

Submitted to
THE UNITED STATES ATOMIC ENERGY COMMISSION
by
THE BABCOCK & WILCOX COMPANY
Atomic Energy Division
Lynchburg, Virginia

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1. INTRODUCTION

In March 1967 The Babcock & Wilcox Company (B&W) initiated a program of analytical methods development applicable to the physics design of fast breeder reactors. One phase of this effort has as its goal the development and utilization of a design-oriented, one-dimensional criticality and depletion code applicable to the physics analysis of fast reactors. This phase of the methods development effort is under full sponsorship of the U. S. Atomic Energy Commission (AEC). Utilization of the code will consist of an investigation in which the design code FARED will be used to relate the sensitivity of certain fast reactor design parameters to the complexity of the physics model used to evaluate them. Finally, several fast reactor critical experiments will be analyzed to verify the applicability of the code and its cross-section libraries. Future topics of study under the FARED project include (1) extension of the criticality and depletion segments to two dimensions with provision for complex fuel management studies, and (2) study of the importance of generating multigroup cross sections using space-dependent spectrum weighting.

2. STATEMENT OF OBJECTIVES

2.1. ENDF/B Implementation

The most convenient and reliable source for much of the cross-section and resonance parameter data required by the FARED code is the Evaluated Nuclear File Project which is sponsored by the AEC and directed by the Brookhaven National Laboratory.¹ The ENDF/B data tape, its service routines, and the fast reactor group constant codes are the components of major interest to the FARED project. The MC² code is a particularly versatile group constants code,² and, although it is unsuitable for routine design application,* transfer matrices and certain smooth cross-section data prepared by this code can be used to construct a FARED library tape. For example, a single FARED library tape could be prepared for use in the design of sodium-cooled reactors by merging the results from MC² for a typical core and a typical blanket composition. The fine group transfer matrices and non-resonance cross-section data would be assumed valid for use in a wide variety of sodium-cooled reactor design studies. Additional library tapes could be prepared for use in the design of steam-cooled or gas-cooled reactors. Furthermore, during a criticality or depletion study, the FARED code may access either one of two library tapes. Thus, an accurate criticality determination of the reactivity worth of coolant in a particular region of the reactor may be made at any time step if one of the library tapes has been prepared for a typical composition with coolant deleted.

The specific objectives of the ENDF/B implementation phase of the FARED project are as follows:

- - - - -
*For a single composition of practical interest, MC² requires approximately 80 minutes of CDC-3600 machine time.³

1. To perform the compilation, debugging, and testing necessary to bring the ENDF/B service routines (CRECT, DAMMET, EDIT), the ETOE code,⁴ and the MC² code into full production use.*

2. To make appropriate modifications to these ENDF/B codes to facilitate the preparation of FARED library tapes.

2.2. FARED Code Development

The FARED code is designed to perform rapid and accurate one-dimensional criticality and depletion calculations for fast reactors. A large number of such calculations are usually performed during the conceptual and preliminary design stages when parametric and optimization studies are required. FARED will facilitate the evaluation of much of the static physics data required for these studies since many important design parameters, such as power profiles, breeding ratios, neutron lifetime, and delayed fractions, Doppler coefficients, and coolant-void coefficients, can be provided automatically with a minimum of user input. The major components of the FARED code are listed here:

1. REGA — A routine that prepares a 26-group (or less) microscopic cross section set for each region (or a combination of regions) in the reactor. REGA performs a fundamental-mode (B_1) spectrum calculation in 100 groups or less for each specified composition; for resonance nuclides, cross sections in the resolved and unresolved resonance ranges are prepared at run time by the RERIP subroutine.

2. RAIM — The RAIM routine accepts the microscopic libraries prepared by REGA, forms macroscopic diffusion parameters for each region, and computes the reactor eigenvalue and space-dependent fluxes using one-dimensional diffusion theory. Criticality searches and edit options are similar to those available in the MACH 1 code.⁵

3. REDEP — This routine performs the reactor burnup calculation using the microscopic libraries prepared by REGA and the fluxes from the RAIM segment. Reactor configuration changes (for fuel management studies) may occur at any time step.

*Implementation is to be made on the CDC-6600 at New York University.

The 26-group microscopic libraries produced by REGA may be regenerated at any time step during a FARED run; additional sets will always be prepared when coolant-void and Doppler coefficient calculations are to be evaluated by a criticality calculation (as opposed to a perturbation calculation, which is also available).

The objectives of the FARED code development effort are as follows:

1. To formulate, code, and production test a one-dimensional criticality and depletion code for use in performing rapid and accurate fast breeder reactor static physics studies.
2. To provide carefully evaluated nuclear data libraries as required for utilization of the FARED code.

2.3. Static Physics Analysis

As a result of the many options available in FARED, the applicability of the code is not limited to any specific reactor type; sodium-, steam-, or gas-cooled, annular, cylindrical, or modular assemblies can be studied using a variety of physics models. This flexibility makes FARED ideal for performing sensitivity and parametric studies which can be used to select optimum physics models for analyzing the various fast reactor concepts. Further, the calculational and edit options available in FARED will also facilitate the analysis of fast reactor critical experiments, particularly when a homogeneous model can be specified. These studies will aid in determining the general applicability of the code and its cross-section libraries.

The objectives of the statics physics analysis effort are as follows:

1. To perform a series of carefully planned computer "experiments" in which the FARED code and, in a few cases, other existing design codes are used to investigate various physics models that have been or could be employed in designing fast breeder reactors.
2. To perform an analysis of a limited number* of fast critical experiments in order to test the criticality segments of FARED, including the cross-section libraries used by the REGA routine.

*Generally restricted to soft-spectrum, plutonium-fueled criticals.

3. ENDF/B IMPLEMENTATION (A. Z. Livolsi, K. E. Grimes)

Implementation of the entire ENDF/B data processing complex is being performed on the CDC-6600 computer at New York University's Courant Institute of Mathematical Sciences (CIMS). Source decks for each of the programs discussed in this section have been placed on the CIMS master tape, from which they can be called for modification, compilation, and execution using a few control statements.

3.1. Service Routines

CRECT — A code to correct an ENDF/B data tape in BCD format. The code will correct records or fields of records; delete materials, files, or sections; or insert records.

DAMMET — A code to delete, alter, and/or merge ENDF/B tapes in either BCD or binary mode. The output tape may be a standard or an alternate arrangement, a binary or a BCD tape.

EDIT — A code to edit data tapes in binary mode, presenting the data in useful table form.

These service routines are now fully operational on the CDC-6600 and have been used to correct ENDF/B for erroneous entries, to merge ENDF/B tapes, and to prepare a mode 2 version of an ENDF/B tape for use by the ETOE code.

3.2. The ETOE Code

The ETOE code accepts an input mode 2 ENDF/B tape and prepares a complete cross-section data tape for use by the MC² code. The overlay structure of the original version of this code has been replaced, so that the code consists of a main program and several subroutines; the entire program is loaded into the core at run time. The sample case distributed with the program has been successfully executed, and ENDF/B tape 102 has been processed by ETOE to form an MC² library containing sodium, iron, ²³⁹Pu, and ²³⁸U. ETOE appears to be fully operational.

3.3. The MC² Code

The MC² code is an unusually versatile cross-section averaging program for use in computing group constants for fast reactors. The overlay structure of the original code has been replaced by a chain and subroutine structure,* and common storage allocation and certain logic statements have been modified according to the suggestions of A. Aronson of Brookhaven National Laboratory. Using an MC² library prepared by Argonne National Laboratory, the sample case distributed with the code was executed successfully. A more realistic test problem was attempted using an MC² library prepared by ETOE from ENDF/B tape 102. This problem is failing in chain 4 (Legendre coefficient preparation) because of erroneous and/or incompatible entries on the ENDF/B tape; there appears to be no difficulty with the MC² code itself. New ENDF/B tapes (114, 115) have been released containing data that have been examined thoroughly and are judged essentially free of erroneous entries. A new ETOE run will be made using a tape formed by merging ENDF/B tapes 114 and 115, and attempts to execute a practical MC² test problem will continue.

A special version of the MC² code has been prepared to permit the editing of group constants for up to 68 groups. This feature will facilitate the preparation of library tapes for the FARED code. Tests of this expanded version of MC² will be made when the original code has been successfully executed.

- - - - -
*This procedure is necessary since N. Y. U. uses the first operating system devised for the CDC-6600, viz., the CHIPPEWA system. Although it is an old operating system, considerable time during the months of August and September was spent debugging the system's Fortran compiler.

4. FARED CODE DEVELOPMENT

(D. H. Roy, J. Helholtz)

The major components of the FARED code and the main lines of flow are sketched in Figure 4-1. The cross-section sets formed by REGA are used by the RAIM diffusion theory routine to determine the reactor flux profile in each of 26 (or less) energy groups. The cross sections and fluxes are then used by the REDEP routine to perform pointwise depletion in each energy group. FARED will be coded in order to maximize the number of library sets which may be used by RAIM and yet maintain running times consistent with the design nature of the code.

4.1. REGA — The Group Constants Generator

4.1.1. Library Tape

A computer program, RETAP, will be written to construct, modify, and/or edit a REGA library tape.* Most of the information required on the library tape can be obtained from ENDF/B tapes and MC² output; data that cannot be obtained from these sources include (1) delayed neutron data, (2) depletion chain information, (3) energy yield per fission for fissile nuclides, and (4) certain source spectra.

The structure of the REGA library tape is given below. It is a design-oriented compilation in that the tape contents have been selected so as to minimize user input and to provide for the editing of essentially all important nuclear design parameters.

* Eventually, this program will require as data input only an ENDF/B tape. An MC²-type calculation will be performed to produce transfer matrices and non-resonance cross sections, and the REGA library will be produced directly.

Figure 4-1. Major FARED Components

Table 4-1. Structure of the REGA Library

Item No.	Format	Description
1.	716	NTLIB, NHED, NGRP, NMAT, NSPEC, NFP, NDCH NTLIB = Numeric tape label NHED = No. of header cards describing this tape NGRP = No. of microgroups for which tape is written (≤ 100) NMAT = No. of materials (nuclides) on this tape NSPEC = No. of source spectra NFP = No. of fission product nuclides NSC = No. of standard depletion chains
2.	12A6	NHED cards of alphanumeric tape description
3.	6E12.6	(EGRP(I), I = 1, NGRP + 1) EGRP(I) = Microgroup energy bounds
4.	<u>For Each Source Spectrum Give</u>	
4.1.	16, 11A6	SID, Alphanumeric description of source
4.2.	6E12.6	(S(I), I = 1, NGRP) SID = Numeric source ID S(I) = Source spectrum
5.	<u>For Each Standard Depletion Chain Give</u>	
5.1.	2I6, 10A6	LCHAIN, NOC, Alphanumeric description of chain LCHAIN = Chain ID NOC = No. of nuclides in this chain
	<u>For Each Nuclide in the Chain Give</u>	
5.2.	12I6	NCID, NSOR, (IDS(I), MSR(I), I = 1, NSOR) NCID = Nuclide ID

Table 4-1 (Cont'd)

<u>Item No.</u>	<u>Format</u>	<u>Description</u>
		NSOR = No. of sources (parents) for formation of this nuclide
		IDS = ID of source nuclide (ignore for fission)
		MSR = Reaction type by which IDS → NCID
		= 1 for fission
		= 2 for capture
		= 3 for radioactive decay
		= 4 for n, 2n

6. For Each Nuclide on this Tape Give

6.1.	7I6	(ISOP(I), I = 1, 7)
		ISOP(1) = 0 fissile
		= 1 non-fissile
		ISOP(2) = 0 no n, 2n xsections
		= 1 n, 2n xsections given
		ISOP(3) = 0 no ϵ_j^l data [$\sigma(n, p) + \sigma(n, \alpha)$]
		= 1 ϵ_j^l data given
		ISOP(4) = -1 non-resonance nuclide
		= 0 resolved resonance data given
		= 1 resolved and unresolved data given
		ISOP(5) = 0 no delayed neutron data
		= 1 delayed neutron data given
		ISOP(6) = 0 no depletion data
		= 1 depletion data given
		ISOP(7) = No. of downscatter groups in transfer matrix

If ISOP(5) = 1, Give

6.2.	6E12.6	(DLAMB(I), I = 1, 6)
		DLAMB(I) = Decay constant for de- layed group I

Table 4-1 (Cont'd)

<u>Item No.</u>	<u>Format</u>	<u>Description</u>
6.3.	6E12.6	(DBETA(I), I = 1, 6) DBETA(I) = Delayed fraction for each group I
6.4.	6E12.6	(DSPEC(I), I = 1, NGRP) DSPEC(I) = Delayed fission spectrum for this set
<u>If ISOP(6) = 1, Give</u>		
6.5.	5(I6, E12.6)	(DCON, EFIS, (IDFP(J), YFP(J), J = 1, NFP)) DCON = Decay constant for this nuclide EFIS = Energy release per fission (Mev/fission) IDFP = Fission produce nuclide ID YFP = Yield of this fission product (IDFP) Only if ISOP(1) = 1
<u>If ISOP(4) = 0, 1, Give Resolved Resonance Data</u>		
6.6.	2I6, 3E12.6	(NECOR, NPKS, AM, SPIN, R) NECOR = No. of energy points at which smooth data are given NPKS = No. of resolved peaks AM = Nuclear mass SPIN = Nuclear spin, I R = Scattering length
<u>For Each Peak Give (1 card/peak)</u>		
6.7.	6E12.6	(EZERO, GAMN, GAMG, GAMF, ABN, AJ) EZERO = Peak energy GAMN = Neutron width GAMG = Capture width GAMF = Fission width

Table 4-1 (Cont'd)

<u>Item No.</u>	<u>Format</u>	<u>Description</u>
		ABN = Abundance of isotope to which this peak belongs
		AJ = Spin state of compound nucleus
<u>After Peak Data Give</u>		
6.8.	6E12.6	(ECOR(I), I = 1, NECOR)
		ECOR(I) = Energy points at which smooth data are given
6.9.	6E12.6	(RCAP(I), I = 1, NECOR)
		RCAP(I) = Smooth capture xsections
<u>If ISOP(1) = 1, Also Give</u>		
6.10.	6E12.6	(RFIS(I), I = 1, NECOR)
		RFIS(I) = Smooth fission xsections
<u>If ISOP(4) = 1, Give Unresolved Resonance Data</u>		
6.11.	2I6, E12.6	(NJSO, NJS1, ECUR)
		NJSO = No. of J states for $l = 0$
		NJS1 = No. of J states for $l = 1$
		ECUR = Upper bound of unresolved region
<u>For Each J State Give</u>		
6.12.	3I6, 3E12.6	(NRN, NEF, MU, BJ, GG, D)
		NRN = No. of energy points at which reduced neutron width is given
		NEF = No. of energy points at which fission width is given
		MU = No. of exit channels for this state
		BJ = J for this state
		GG = Mean capture width
		D = Mean level spacing

Table 4-1 (Cont'd)

<u>Item No.</u>	<u>Format</u>	<u>Description</u>
6.13.	6E12.6	(EFN(I), GNZ(I), I = 1, NRN) EFN(I) = Energy point, ev GNZ(I) = Reduced neutron width at this point
6.14.	6E12.6	(EFF(I), GF(I), I = 1, NEF) EFF(I) = Energy point GF(I) = Fission width at this point
<u>Finally, Give Microgroup Xsection Data</u>		
6.18.	6E12.6	(XCAP(I), I = 1, NGRP) XCAP(I) = Capture xsection for Group I
6.19.	6E12.6	(XFIS(I), I = 1, NGRP) XFIS(I) = Fission xsection for Group I [Only if ISOP(1) = 1]
6.20.	6E12.6	(NU(I), I = 1, NGRP) FNU(I) = No. neutrons per fission for Group I [Only if ISOP(1) = 1]
6.21.	6E12.6	(XN2N(I), I = 1, NGRP) XN2N(I) = N, 2N xsection for Group I [Only if ISOP(2) = 1]
6.22.	6E12.6	(XEP(I), I = 1, NGRP) $XEP(I) = \sigma_{n,p}^I + \sigma_{n,\alpha}^I$ for Group I [Only if ISOP(3) = 1]
6.23.	6E12.6	((XTRO(I,J), I = 1, NGRP), J = I, MIN(I + NDS, 69)) Isotropic transfer matrix NDS = ISOP(7)

Table 4-1 (Cont'd)

<u>Item No.</u>	<u>Format</u>	<u>Description</u>
6.24.	6E12.6	((XTR1(I, J), I = 1, NGRP), J = I, MIN (I + NDS, 69))
		Anisotropic transfer matrix

4.1.2. Preparation of Microscopic Cross Section Sets for RAIM

The procedure for computing broad group microscopic cross sections in FARED is similar to that used in the GAM code.⁶ For a given homogeneous composition, REGA performs a fundamental-mode, B₁ computation of the flux and current in each group (the NGRP groups specified on the REGA library tape) and uses these as weighting functions to form cross sections for the broad group structure (≤ 26). The fundamental-mode flux and current are given by^{7, 8}

$$C_{1,j} \phi_j + C_{2,j} J_j = S_j + C_{3,j} \quad (4-1)$$

$$-\frac{C_2}{3} \phi_j + C_{4,j} J_j = C_{5,j} \quad (4-2)$$

where

$$C_{1,j} = \sum_{\ell} N_{\ell} (\sigma_{T,j}^{\ell} - \sigma_{j \rightarrow j}^{\ell,0}) \quad (4-3)$$

where

N_{ℓ} = homogeneous number density of element ℓ

$$\sigma_{T,j}^{\ell} = \sigma_{c,j}^{\ell} + \sigma_{f,j}^{\ell} + \epsilon_j^{\ell} - \sigma_{2n,j}^{\ell} + \sum_{k \geq j} \sigma_{j \rightarrow k}^{\ell,0}$$

$$\begin{aligned} C_{2,j} &= -\sqrt{|B^2|} \quad \text{if input } B^2 < 0 \\ &= \sqrt{B^2} \quad \text{if input } B^2 \geq 0 \end{aligned} \quad (4-4)$$

$$C_{3,j} = \sum_{i < j} \phi_i \sum_{\ell} N_{\ell} \sigma_{i \rightarrow j}^{\ell, 0} \quad (4-5)$$

$$C_{4,j} = \sum_{\ell} N_{\ell} \left(\gamma_j \sigma_{T,j}^{\ell} - \sigma_{j \rightarrow j}^{\ell, 1} \right) \quad (4-6)$$

$\gamma_j = 1$ if input $B^2 > 0$ or P1 option specified

$$= \frac{\gamma_j \tan^{-1} \gamma_j}{3 \left(1 - \frac{1}{\gamma_j} \tan^{-1} \gamma_j \right)} \quad \text{if } B_1 \text{ option specified and } B^2 > 0.$$

$$= \frac{\gamma_j \tanh^{-1} \gamma_j}{3 \left(\frac{1}{\gamma_j} \tanh^{-1} \gamma_j - 1 \right)} \quad \text{if } B_1 \text{ option specified and } B^2 < 0.$$

$$\gamma_j = \frac{\sqrt{1 B^2 1}}{\Sigma_{Tj}}$$

$$\Sigma_{Tj} = \sum_{\ell} N_{\ell} \sigma_{T,j}^{\ell}$$

$$C_{5,j} = \sum_{i < j} J_i \sum_{\ell} N_{\ell} \sigma_{i \rightarrow j}^{\ell, 1} \quad (4-7)$$

If the B_1 option is specified with $B^2 < 0$ and $\gamma_j > 1.0$, an error message is provided off-line, and the composition is reprocessed using the P_1 option. The functions ϕ_j and J_j are the group flux and current. Solution of the spectrum equations begins in group $j = 1$ (highest energy group) where $C_{3,j} = C_{5,j} = 0$.

For each nuclide in the composition, REGA prepares the following broad group microscopic cross sections for use by the RAIM and REDEP routines

$$\sigma_{c,J}^{\ell} = \frac{1}{\Phi_J} \sum_{j \in J} \phi_j \sigma_c^{\ell, j} \quad (\text{capture}) \quad (4-8)$$

$$\sigma_{f,J}^{\ell} = \frac{1}{\Phi_J} \sum_{j \in J} \phi_j \sigma_{f,j}^{\ell,j} \quad (\text{fission}) \quad (4-9)$$

$$\nu_J^{\ell} = \frac{1}{\Phi_J \sigma_{f,J}^{\ell}} \sum_{j \in J} \phi_j \nu_j^{\ell} \sigma_{f,j}^{\ell,j} \quad (\text{neutrons/fission}) \quad (4-10)$$

$$\sigma_{K \rightarrow J}^{\ell} = \frac{1}{\Phi_K} \sum_{k \in K} \sum_{j \in J} \phi_k \sigma_{k \rightarrow j}^{\ell,0} \quad (\text{transfer matrix}) \quad (4-11)$$

(K ≠ J)

$$\sigma_{tr,J}^{\ell} = \frac{1}{J_J} \sum_{j \in J} J_j \sigma_{tr,j}^{\ell,j} \quad (\text{transport}) \quad (4-12)$$

$$\sigma_{tr}^{\ell,j} = \sigma_T^{\ell,j} - \sum_{k \geq j} \sigma_{j \rightarrow k}^{\ell,1}$$

$$\chi_J = \sum_{j \in J} S_j \quad (\text{source}) \quad (4-13)$$

$$\sigma_{2n,J}^{\ell} = \frac{1}{\Phi_J} \sum_{j \in J} \phi_j \sigma_{2n,j}^{\ell,j} \quad (n, 2n) \quad (4-14)$$

$$\delta_J^{\ell} = \frac{1}{\Phi_J} \sum_{j \in J} \phi_j (\epsilon_j^{\ell} - \sigma_{2n,j}^{\ell,j}) \quad (\text{removal correction}) \quad (4-15)$$

where

$$\Phi_J = \sum_{j \in J} \phi_j$$

$$J_J = \sum_{j \in J} J_j$$

ℓ = element index

j = microgroup (REGA) index

J = broad group index

The δ_J^{ℓ} term will be used to correct the removal rates in the RAIM diffusion calculation for the (n, p) and (n, α) reactions and to eliminate the imbalance resulting from the fact that the n, 2n matrix elements sum to

and is evaluated by the STRIP method as described below. For the unresolved portion of the group, the average resonance cross section is given by

$$\bar{\sigma}_{xr}^{m, \text{unres}} = \bar{\sigma}_{xr}^m \left(\frac{ECR + E_{k-1}}{2} \right) \quad (4-20)$$

Assuming that $\phi(E) \propto E^{-1/2}$, we may express $\bar{\sigma}_{xr}^{m, k}$ as

$$\bar{\sigma}_{xr}^{m, k} = \frac{1}{(\sqrt{E_{k-1}} - \sqrt{E_k})} \left[\bar{\sigma}_{xr}^{m, \text{res}} (\sqrt{ECR} - \sqrt{E_k}) + \bar{\sigma}_{xr}^{m, \text{unres}} (\sqrt{E_{k-1}} - \sqrt{ECR}) \right] \quad (4-21)$$

2. The second method for evaluating the effective unresolved resonance cross section is provided for use with relatively coarse micro-group structures. With this option, $\bar{\sigma}_{xr}^m(E)$ is first evaluated at each group boundary in the unresolved energy range and at M points interior to each group, where M is an input integer. Between any two points, E_n and E_{n+1} , the cross section is assumed to be given by

$$\bar{\sigma}_{xr}(E) = \bar{\sigma}_{xr}(E_n) + A_n \ln \frac{E}{E_n} \quad (4-22)$$

where

$$A_n = \frac{\bar{\sigma}_{xr}(E_{n+1}) - \bar{\sigma}_{xr}(E_n)}{\ln \frac{E_{n+1}}{E_n}}$$

Assuming that $\phi(E) \propto E^{-1/2}$, the average group cross section is then given by

$$\bar{\sigma}_{xr}^{m, j} = \frac{1}{(\sqrt{E_{j-1}} - \sqrt{E_j})} \sum_{n \in j} \left\{ A_n (\sqrt{E_{n+1}} \ln E_{n+1} - \sqrt{E_n} \ln E_n) + [\bar{\sigma}_{xr}^m(E_n) - A_n (\ln E_n + 2)] (\sqrt{E_{n+1}} - \sqrt{E_n}) \right\} \quad (4-23)$$

where the sum on n refers to the $M + 1$ intervals in the range $E_j \leq E \leq E_{j-1}$. For the group (k) containing ECR, the average group cross section is computed as in part 1 if

$$\ln \frac{E_{k-1}}{ECR} \leq 0.25$$

If the lethargy width of the unresolved portion is greater than 0.25, equation 4-1 is used with $\bar{\sigma}_{xr}^{m, \text{unres}}$ evaluated by 4-23 with $E_j = ECR$ and $E_{j-1} = E_{k-1}$.

Evaluation of the unresolved resonance cross section for each ℓ, J sequence, $\bar{\sigma}_{xr, \ell, J}^{in}(E)$, is performed in a manner similar to that used in the MC² code. Overlap and interference are ignored and the NR approximation is assumed valid so that

$$\bar{\sigma}_{cr, \ell, J}^m(E) = \frac{\sigma_b \bar{\Gamma}_Y^{\mu_{\ell, J}}}{\bar{D}_{\ell, J}} \langle J \rangle_n \quad (\text{for non-fissile nuclides}) \quad (4-24)$$

$$= \frac{\sigma_b \bar{\Gamma}_Y^{\mu_{\ell, J}}}{\bar{D}_{\ell, J}} \langle J \rangle_{n, f} \quad (\text{for fissile nuclides}) \quad (4-25)$$

$$\bar{\sigma}_{fr, \ell, J}^m(E) = \frac{\sigma_b \bar{\Gamma}_f^{\mu_{\ell, J}}}{\bar{D}_{\ell, J}} \langle \Gamma_f J \rangle_{n, f} \quad (4-26)$$

where

$$\langle J \rangle_n = \frac{1}{\sqrt{\pi}} \int_0^\infty \frac{e^{-x}}{\sqrt{x}} J[\theta(x, 0), \beta(x, 0)] dx$$

$$\langle J \rangle_{n, f} = \frac{1}{\sqrt{\pi}} \int_0^\infty e^{-y} dy \int_0^\infty \frac{J[\theta(x, y), \beta(x, y)]}{\sqrt{x}} dx$$

$$\langle \Gamma_f J \rangle_{n, f} = \frac{1}{\sqrt{\pi}} \int_0^\infty y e^{-y} dy \int_0^\infty \frac{J[\theta(x, y), \beta(x, y)]}{\sqrt{x}} dx$$

$$\theta(x, y) = \left(\frac{A}{4kT} \right)^{1/2} \left(\frac{\bar{\Gamma}_Y}{\sqrt{E}} + \frac{y \bar{\Gamma}_f}{\sqrt{E}} + 2x \langle \Gamma_n^0 \rangle \right)$$

$$\beta(x, y) = \frac{\sigma_b}{2.6038 \cdot 10^6 \cdot g_j} \left[\frac{\sqrt{E}}{2x \langle \Gamma_n^0 \rangle} (\bar{\Gamma}_Y + y \bar{\Gamma}_f) + E \right]$$

The integral terms in these expressions represent averages of the Doppler broadening function, $J(\theta, \beta)$, over χ -squared distributions of one ($x = \Gamma_n / \langle \Gamma_n^0 \rangle$) and two ($y = \Gamma_f / \overline{\Gamma}_f$) degrees of freedom. These averages are formed by the JAVNF subroutine which uses a 12-point Laguerre-Gauss quadrature to evaluate the integrals. The number of integration points was selected on the basis of a study designed to optimize the integration procedure; this investigation is described in Appendix B. Since the Doppler broadening function will appear 144 times during a single calculation of $\overline{\sigma}_{xr, \ell, J}^m$ for a fissile nuclide, an additional research effort was conducted in order to devise an extremely rapid, yet accurate, method for evaluating this function. This study resulted in a function routine, FJINT, which returns a value of $J(\theta, \beta)$ correct to 0.1% or better in 2 milliseconds on the Philco 2000/211. This routine is described in Appendix A. In equations 4-25 and 4-26, the average reduced neutron width, $\langle \Gamma_n^0 \rangle$, and the average fission width, $\overline{\Gamma}_f$, at any energy point E are obtained from the GNZ and GF listings on the REGA library tape; linear interpolation is assumed. Other parameters appearing in these equations are

$\mu_{\ell, J}$ = No. of entrance channels for this ℓ, J sequence

T = temperature, °K

k = Boltzmann constant = 8.61706×10^{-5} eV/°K

$\overline{D}_{\ell, J}$ = mean level spacing

$\overline{\Gamma}_\gamma$ = average capture width

$$g_J = \frac{1}{2} \left(\frac{2J+1}{2I+1} \right)$$

I = spin of the target nucleus

The parameter σ_b is the effective scattering cross section per absorber atom; if the input lump dimension is zero,

$$\sigma_b = \sigma_{pr} + \frac{1}{N_r^{HOM}} \sum_{i \neq r} N_i^{HOM} \sigma_{pi}$$

where

$$\begin{aligned}\sigma_{pi} &= \text{potential scattering cross section for nuclide } i \\ &= 4\pi R_i^2\end{aligned}$$

If the lump dimension is non-zero,

$$\sigma_b = \sigma_{pr} + \frac{1}{N_r^{\text{lump}}} \sum_{i \neq r} N_i^{\text{lump}} \sigma_{pi} + \frac{1.35(1 - C)}{N_r^{\text{lump}} \bar{l}(1 + 0.35C)}$$

where

C = Dancoff factor

$\bar{l}$ = mean chord length of the lump

The Dancoff factor and mean chord length may be supplied by the user or computed by the program; the program computation for C is performed by the DANCEF routine and is available only for plate or pin lattices. The DANCEF routine uses a modification of Sauer's method as described in reference 9.

Effective cross sections for the resolved resonance region are computed using a method identical to that employed in the STRIP code.¹⁰ The STRIP calculation begins at the highest $(ECR)^m$ among all the resonance nuclides; ECR is defined by equation 4-16. The entire resolved resonance region, extending from $(ECR)^m_{MAX}$ to the lowest bound of the last REGA microgroup, is then overlaid with a fine energy mesh as described in section 3.2 of reference 10. For the FARED code, however, all parameters used in the mesh width calculation ($\beta_0, \beta_\infty, d_0, d_\infty, h$, and J) are fixed; an investigation now in progress will optimize the selection of these parameters for fast reactor calculations. In the vicinity of each ECR, the width of an interval is adjusted somewhat in order that $(ECR)^m$ for all m correspond to a boundary of one of the fine mesh intervals. For energies greater than $(ECR)^m_{MAX}$, FARED assumes the flux to be proportional to $E^{-1/2}$.

The resolved resonance calculation in REGA may be performed for either a homogeneous mixture of nuclides or for a two-region plate or pin cell, each region of which may contain resonance nuclides. For the two-region cell, the flux in each region is obtained from

$$\begin{aligned}
(\Sigma_{T1}^r \Delta_r - P_{11}^r A_1^r) \phi_1^r - P_{21}^r V_2 A_2^r \phi_2^r &= P_{11}^r B_1^r + P_{21}^r V_2 B_2^r \\
- P_{12}^r A_1^r \phi_1^r + (\Sigma_{T2}^r \Delta_r - P_{22}^r V_2 A_2^r) \phi_2^r &= P_{12}^r B_1^r + P_{22}^r V_2 B_2^r
\end{aligned} \tag{4-27}$$

where

ϕ_1^r, ϕ_2^r = flux in energy interval r for regions 1 and 2

Σ_{Tj}^r = total macroscopic cross section of region j
at the midpoint of interval r

P_{ij}^r = first flight collision probability described in
section 4 of reference 10

Δ_r = width of mesh interval $r = E_{r-1} - E_r$

V_2 = moderator-to-lump volume ratio

A_j^r, B_j^r = matrix coefficients defined in section 1 of
reference 10 (These coefficients are func-
tions of previously computed fluxes and
cross sections.)

The microscopic total cross section for any material m at mesh point E_r^* , $[E_r^* = 1/2(E_{r-1} + E_r)]$, is composed of the following:

1. The single-level Breit-Wigner fission and capture contributions from the first k peaks above and below the mesh point E_r^* ; k is an input integer.
2. The single-level Breit-Wigner scattering contribution from the first peak above the mesh point and the nearest k peaks below the mesh point. This contribution includes the resonance and interference components.
3. The smooth capture and fission contributions given in the RCAP and RFIS listings on the REGA library tape; linear interpolation is assumed.
4. The potential scattering cross section, σ_{pm} .

If $(ECR)^m < E_r^*$ (that is, if a given mesh point lies above the resolved resonance cutoff for material m), then the contribution this material makes to the total cross section at E_r^* is taken as σ_{pm} .

In a heterogeneous problem, the flux in the region exterior to lump is always evaluated explicitly. When a particular

resonance nuclide appears in both lump and moderator regions, the Dancoff factor is calculated at each mesh point and the effective resonance cross section is given by

$$\sigma_x^{m,j} = \frac{\sum_{r \in j} (N_m^1 \phi_1^r + N_m^2 \phi_2^r V_2) \sigma_x^{m,r}}{(N_m^1 + N_m^2 V_2) \sum_{r \in j} (\phi_1^r + \phi_2^r)} \quad (4-28)$$

4.2. RAIM — The Flux and Eigenvalue Routine

The RAIM diffusion theory routine calculates neutron fluxes in up to 26 energy groups at a maximum of 100 space points. The reactor may be divided into as many as 15 zones in a slab, spherical, or cylindrical configuration. In each reactor zone, RAIM solves a diffusion equation of the form

$$D_i \nabla^2 \phi_i(\vec{r}) + \Sigma_{Ti} \phi_i(\vec{r}) = \frac{\chi_i}{\lambda} S(\vec{r}) + \sum_{j=\text{MAX}(1, i-\text{NDS})}^{i-1} \phi_j(\vec{r}) \Sigma_{j \rightarrow i} \quad (4-29)$$

where NDS = the number of downscatter groups

$$\Sigma_{j \rightarrow i} = \Sigma_{j \rightarrow i}^{\text{inelastic}} + 2 \Sigma_{j \rightarrow i}^{n, 2n} + \Sigma_{j \rightarrow i}^{\text{elastic}}$$

$$S(\vec{r}) = \sum_{\text{all } j} \phi_j(\vec{r}) \nu_j \Sigma_{fj}$$

$$\Sigma_{Ti} = D_i B_{zi}^2 + \Sigma_{ci} + \sum_{j>i} \Sigma_{i \rightarrow j} + \Sigma_{fi} + \Delta_i$$

$$\Delta_i = \sum_{\ell \in \text{region}} N_\ell \delta_i^\ell \quad (\text{see equation 4-15})$$

The various coefficients required are computed for each zone using a microscopic cross-section library for the reactor block containing that zone (a zone and a block may be identical). A microscopic library for the materials in each block may be supplied by the user or generated by the REGA spectrum routine. The number of downscatter groups is limited to 12.

The boundary conditions, criticality searches, and edit options available in RAIM include those available in the MACH-1 code plus a distributed Doppler coefficient calculation and a zone-dependent coolant worth evaluation in which the RAIM calculation is repeated using diffusion parameters characteristic of the voided zone. Work is currently underway to specify the distributed $T dk/dT$ calculation.

4.3. REDEP — The Burnup Segment

The REDEP routine in FARED performs point-wise depletion in each of the RAIM energy groups. The time dictionary in the REDEP routine controls time steps, edits, and reactor configuration changes associated with fuel management studies; the time dictionary is accessible to the user. It is expected that when a depletion run is designated, most of the fission product and decay chain data required will be determined by analysis of the input nuclide identifications and proper nuclide number densities initialized automatically. It may be necessary for the user to supply a depletion assignment parameter for each nuclide in the $t = 0$ mixes. For a given nuclide, the following situations may arise:

1. The nuclide could be non-depletable:

$$\frac{dN_\ell}{dt} = 0$$

2. The nuclide could be pure depletable:

$$\frac{dN_\ell}{dt} = - \sum_j \phi_j N_\ell \sigma_{a,j}^\ell \quad (j \equiv \text{group index})$$

3. The nuclide could be a member of a standard chain described on the REGA library tape:

$$\frac{dN_\ell}{dt} = \sum_i q_i - (\lambda_\ell + \sum_j \phi_j \sigma_{a,j}^\ell) N_\ell$$

$$i = 1, 2, \dots, \text{NSOR}$$

$$\begin{aligned}
q_i &= \sum_j \phi_j \sum_m N_m \gamma_m^\ell \sigma_{f,j}^m & \text{if MSR} = 1 \\
&= \sum_j \phi_j N_{IDS} \sigma_{c,j}^{IDS} & \text{if MSR} = 2 \\
&= \lambda_{IDS} N_{IDS} & \text{if MSR} = 3 \\
&= \sum_j \phi_j N_{IDS} \sigma_{2n,j}^{IDS} & \text{if MSR} = 4
\end{aligned}$$

4. The nuclide could be a fission product:

$$\frac{dN_\ell}{dt} = \sum_j \phi_j \sum_m N_m \gamma_m^\ell \sigma_{f,j}^m - (\lambda_\ell + \sum_j \phi_j \sigma_{a,j}^\ell) N_\ell$$

5. The nuclide could be simply radioactive:

$$\frac{dN_\ell}{dt} = -(\lambda_\ell + \sum_j \phi_j \sigma_{a,j}^\ell) N_\ell$$

It may be most efficient to require a standard chain assignment for every input isotope where the assignments include any of the cases listed in 1 through 5 above. In any event, FARED will be able to construct the appropriate depletion equations from little more than the input nuclide specifications.

Detailed specifications of the REDEP will begin in February. Special studies will be conducted to (1) determine a method for solving the depletion equations (including Δt specification) which is optimum for fast reactor design; and (2) formulate reactor configuration change options which will facilitate realistic fast breeder reactor fuel management studies.

5. STATIC PHYSICS ANALYSIS

5.1. Sensitivity Studies

(These studies are to be performed during the Task 2 effort and will not begin until the entire FARED code is completed.)

5.2. Analysis of Criticals

(These studies are to be performed upon completion of the criticality segment of FARED.)

APPENDIX A
Calculation of the Doppler-Broadening Function

A fast calculation of the Doppler-broadening function, $J(\theta, \beta)$, is of vital importance to the FARED code since the function will be required many times during the evaluation of unresolved resonance cross sections for fissile materials. Existing methods^{4,11} for calculating the function were found unsatisfactory and a new method that will compute J rapidly (~2 milliseconds) and accurately (0.1% or better) for a wide range of θ and β values has been devised.

The Doppler-broadening function is defined as

$$J(\theta, \beta) = \int_0^{\infty} \frac{\psi(\theta, x)}{\psi(\theta, x) + \beta} dx \quad (A-1)$$

where $\psi(\theta, x)$ is the Doppler-broadened line shape function and the other parameters are given in section 4.1.3. In the ERIC 2 code,¹¹ the integration in equation A-1 is carried out numerically for $0 \leq x \leq 1000$ using 64-point Gauss quadrature; the remainder is obtained analytically assuming the natural line shape for ψ for $x > 1000$. Besides being very slow, the ERIC 2 method is highly inaccurate* for small values of θ and β .

The MC² code improves the ERIC 2 treatment by first expressing ψ as

$$\psi(\theta, x) = \sqrt{\pi} \frac{\theta}{2} u\left(\frac{x\theta}{2}, \frac{\theta}{2}\right)$$

where u is the real part of the complex probability integral

$$w(z) = u(z) + iv(z) = \frac{i}{\pi} \int_{-\infty}^{\infty} \frac{e^{-t^2}}{z - t} dt$$

and then rewriting equation A-1 as

*As a matter of fact, J cannot exceed a certain limit (~1000) for any combination of θ and β .

$$J(\theta, \beta) = \int_0^y \frac{u\left(y, \frac{\theta}{2}\right) dy}{\frac{\theta}{2} u\left(y, \frac{\theta}{2}\right) + \frac{\beta}{\sqrt{\pi}}} + \Delta J \quad (\text{A-2})$$

where

$$y = \frac{x\theta}{2}$$

ΔJ = remainder

$$= \int_{2y/\theta}^{\infty} \frac{\psi(\theta, x)}{\psi(\theta, x) + \beta} dy \quad (\text{A-3})$$

In the MC² code $y = 100$ and the integral in equation A-2 is evaluated using 32-point Gauss quadrature; the remainder, ΔJ , is again evaluated by assuming the natural line shape for ψ . Although the accuracy of this method is probably better than that of the ERIC 2 program, the running time is far from satisfactory.

To improve the overall calculation of J in both accuracy and speed of evaluation, the first step was to improve the calculation of ΔJ . This was achieved by including an additional term in the "near-natural" shape¹² of ψ in equation A-3:

$$\psi(\theta, x) \approx \frac{1}{1+x^2} \left[1 + \frac{2(3x^2-1)}{(1+x^2)^2\theta^2} \right] \quad (\text{A-4})$$

Using this substitution and recognizing that the product $x\theta$ is a large number, equation A-3 yields

$$\begin{aligned} \Delta J = \arctan \left(\frac{1}{R} \sqrt{\frac{\beta+1}{\beta}} \right) & \left(1 + \frac{\beta}{\theta^2} \frac{8\beta^2 + 12\beta + 3}{\beta+1} \right) \\ & - \frac{2\beta}{\theta^2} \left(4 \arctan \frac{1}{R} + \frac{2\beta + \frac{3}{2}}{\beta+1} \frac{R}{\beta+1 + \beta R^2} \right) \end{aligned} \quad (\text{A-5})$$

where

$$R = \frac{2y}{\theta}$$

The primary objective of this portion of the investigation was to devise a method for computing J to an accuracy of 0.1% for all θ and β in the range of practical interest, i.e., for $10^{-3} < \theta < \infty$, $10^{-5} < \beta < \infty$.

To calculate ΔJ , using equation A-5, to the required accuracy it was found that the minimum possible value of y is $y = 3$. For this low value, evaluation of the integral in equation A-2 can be done with a 6-point Gauss quadrature, provided the points are distributed so that most of them are located at low values of y . This is achieved by using the last half of a 12-point Gauss quadrature. For large θ and β , two different asymptotic formulas for $J(\theta, \beta)$ were derived to supplement the quadrature evaluation of A-5: For large β , the integrand of equation A-1 can be expanded in a power series of ψ and A-1 integrated analytically to form

$$\begin{aligned} J(\theta, \beta) &\approx \frac{\pi}{2\beta} \left[1 - \frac{\sqrt{\pi}}{2\beta} \frac{\theta}{2} u\left(0, \frac{\theta}{\sqrt{2}}\right) \right] \\ &= \pi/2\beta \left[1 - \frac{\sqrt{\pi}}{2\beta} \frac{\theta}{2} e^{\theta^2/2} \operatorname{erfc} \theta/\sqrt{2} \right] \end{aligned} \quad (\text{A-6})$$

which is accurate to 0.1% for $\beta > 40$. For large θ , equations A-3 and A-5 can be combined with $y = 0$ to yield

$$J(\theta, \beta) = \frac{\pi}{2\sqrt{\beta(\beta+1)}} + \frac{4\pi\beta}{\theta^2} \left[\frac{2\beta^2 + 3\beta + 3/4}{2(\beta+1)\sqrt{\beta(\beta+1)}} - 1 \right] \quad (\text{A-7})$$

which is accurate to 0.1% for $\theta > 4$. The method requires 20 milliseconds per J calculation on the Philco 2000/211.

When storage is available, carefully constructed tables and interpolation routines offer a still faster method for computing J . The JFINT function routine in FARED uses a two-variable, six-point interpolation formula to interpolate $\log J$ as a function of $\log \theta$ and $\log \beta$; this procedure gave the smoothest representation of J and so minimized the number of values of J required. A table of $\ln J$ (831 values) was constructed using the method above for

$$\theta = 10^{-3} \cdot 2^L \quad \beta = 10^{-5} \cdot 2^K$$

at the points $K = 0(0.75)23.75$ and $L = 0(0.5)12.5$. This procedure, in combination with the asymptotic expressions given in equations A-6 and A-7, will return a value of J in 2 milliseconds that is accurate to 0.1%.

APPENDIX B
Computation of Unresolved Resonance Integrals

The average capture and fission cross sections in the range of unresolved resonances are proportional to the quantities

$$\langle J \rangle_{nf} \text{ and } \langle \Gamma_f J \rangle_{nf} \quad (B-1)$$

respectively, which are the averages of $J(\theta, \beta)$ and $\Gamma_f J(\theta, \beta)$ over the distributions of the neutron and the fission widths. For non-fissile nuclides, only averaging of the neutron width is required.

The expressions in B-1 are obtained by integrating $J(\theta, \beta)$ or $\Gamma_f J(\theta, \beta)$ over the distributions of Γ_n and Γ_f . Γ_n in this study was assumed to have a χ^2 distribution with one degree of freedom, and Γ_f a χ^2 distribution with two degrees of freedom. These integrations were carried out using a variable number of points of the Laguerre-Gauss quadrature formula for a number of isotopes with known unresolved data listed in Table B-1. These isotopes include a total of 25 different spin states, for which unresolved resonance data were taken from ENDF/B. The averaging procedure for all of these states has been checked for several energies between 0.3 and 50 Kev (for ^{238}U and ^{232}Th only between 4 and 50 Kev), for two different temperatures, and for various amounts of dilution. The dilution may be defined by

$$d = \sigma_b / \sigma_p$$

where σ_p is the potential scattering cross section of the absorber and σ_b is the total potential scattering cross section of the medium, as defined in equation A-1. For a most concentrated absorber, $d = 1$. The dilutions checked in this study were 1, 10, 100, and 1000.* This range covers all cases that may have any practical interest in fast reactors.

For every combination of energy and dilution, the averages (B-1) were computed for all the spin states of Table B-1 using a varying

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* With respect to tungsten, the conclusions drawn below are based only on dilutions of 100 and 1000. For the lower dilutions the errors of the resonance integrals are greater than the indicated 1%, but these dilutions of tungsten are of no importance in fast reactors.

number of Laguerre points. For the non-fissile nuclides, the average of $J(\theta, \beta)$ over the distribution of the neutron widths was computed with up to a maximum of 12 Laguerre points. For the fissile nuclides (^{240}Pu and ^{239}Pu), both averages (B-1) were computed with up to 10 points independently for both the neutron width and the fission width. In addition, this procedure was repeated for two temperatures—300 and 600 K—to check the accuracy of the Doppler effect.

The detailed tables and results, too extensive to be presented here, were carefully checked, and a scheme for the minimum allowable number of Laguerre points was constructed. An accuracy of 1% in the resonance integrals of all the cases investigated is attained by using eight points for the neutron width averaging, for both the fissile and non-fissile materials. The number of points necessary for the fission width averaging depends strongly on the value of the average fission width, $\bar{\Gamma}_f$, relative to the radiation width, Γ_γ . Again, to keep the 1% accuracy in resonance integrals, the following scheme for n_f , the number of points for the fission width averaging, may be used:

$$n_f = 8 \text{ if } \bar{\Gamma}_f > 0.6 \Gamma_\gamma$$

$$n_f = 2 \text{ if } \bar{\Gamma}_f \leq 0.6 \Gamma_\gamma$$

In addition to checking the averages (B-1) with variable numbers of Laguerre points, the differences of these averages between 300 and 600 K were computed. It was discovered, quite surprisingly, that the differences are even less sensitive to the number of points than the averages (B-1) themselves, since, as it turns out, the resonance integrals at different temperatures are strongly correlated with each other when computed by a similar method. As a result, if this scheme of Laguerre points is used, the calculated Doppler effect will also be accurate to 1% in all cases, although it is computed as a difference of two numbers.

The unresolved resonance treatment in the FARED code will probably be applied only to few nuclides, since the very diluted, like tungsten, will be assumed as infinitely diluted. A typical run may include unresolved resonance calculations of only ^{239}Pu , ^{240}Pu , and ^{238}U . With

the use of the very fast subroutine for the J function calculation, such a calculation of FARED (even when the unresolved resonance integrals are to be calculated in 20 energy groups) would require a running time of only seconds or less on the CDC 6600. In view of these considerations, it was decided to arbitrarily increase the number of Laguerre points to 12 for both the neutron width and the fission width averaging. In this way the unresolved resonance calculations will be even more reliable, in exchange for a negligible increase in running time.

Table B-1. Isotopes Used for Checking the Averaging of the Unresolved Resonance Integrals

Isotope	ℓ	J
^{240}Pu	0	1/2
	1	1/2, 3/2
^{239}Pu	0	0, 1
^{238}U	0	1/2
	1	1/2, 3/2
^{232}Th	0	1/2
	1	1/2, 3/2
^{182}W	0	1/2
	1	1/2, 3/2
^{183}W	0	0, 1
	1	0, 1, 2
^{184}W	0	1/2
	1	1/2, 3/2
^{186}W	0	1/2
	1	1/2, 3/2

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Troy, New York 12181
Attn: Erwin Gaerttner
14. Westinghouse Electric Corporation
Post Office Box 355
Pittsburgh, Pennsylvania 15230
Attn: Lee Strawbridge
15. The Babcock & Wilcox Company (28)
Barringer, HS	Minner, DA
Central Files (6)	Paternot, HJ
Colpo, PN	Poor, HH/Alliance
Deuster, RW	Probert, PB/Barberton
Helholtz, J	Roy, DH (2)
Holt, CD	Travis, CC/TRG
Hostetler, DR	Webb, RA
Library (2)	Whitaker, BW
Littrell, LW (2)	Wittkopf, WA
Livolsi, AZ	Wolfe, GE