ANPP CODE DEVELOPMENT PROGRAM PRESSURIZED WATER TASK QUARTERLY PROGRESS REPORT NUMBER ONE

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November 1959

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ANPP CODE DEVELOPMENT PROGRAM
PRESSURIZED WATER TASK
QUARTERLY PROGRESS REPORT NUMBER ONE

November 1959

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FOREWORD

This report is submitted by the Nuclear Division of The Martin Company to the U.S. Atomic Energy Commission in compliance with Contract AT(30-1)-2431. The report describes progress from August 1 through October 31, 1959 on the ANPP Code Development Program, Pressurized Water Task.

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SUMMARY

Progress for the first quarter on the ANPP Code Development Program, Pressurized Water Task, is reported.

The BENIKO and RSP physical models for calculating buckling and reflector savings coupling coefficients for synthesis were revised, extended and combined for improved accuracy, flexibility and convenience under the new name, BRAVE.

Program C was modified to eliminate the assumption of the validity of using a bare core slowing-down spectrum and to correct for inelastic scattering in stainless steel, and renamed Program M.

Program F was modified to calculate adjoint fluxes, to include extra lattice intervals and to include the option of spherical geometry, and was renamed Program T.

Characteristics and current status of 15 machine codes of interest are presented.

The details of automating the BRAVE, M and T calculations were developed. The resulting linkage of these three programs is called SYNFar, for "Synthesized Flux and Reactivity Calculation."

The Nuclear FORTRAN Library Tape System was revised for increased efficiency and ease of operation.

A FORTRAN subroutine, called CHANGE, for making efficient changes in a few of the input data of a case without reading in all of the data, was written and checked out.

A FORTRAN subroutine for changing the tape unit numbers used in an IBM-704 FORTRAN program, without reassembly, was written and checked out.

A scheme was developed for reading basic cross-section data from experimental curves using Benson-Lehner equipment and generating a basic cross-section data tape from this information.

A FORTRAN program was outlined and started for generating a multigroup cross-section production data tape from the basic data tape.

A FORTRAN program for applying the Breit-Wigner formula to fill in gaps in the experimental data was written and check out was started.

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Use of the Benson-Lehner equipment to read and convert the basic curvilinear data to punched card form was started for uranium-235.

Critical experiments were performed on two cores during the quarter.

In the first core, critical buckling measurements were performed using gold, thorium and dysprosium foils and the fuel plate scanning technique.

Analytical buckling and reflector savings were compared with the experimental values. The thorium data agreed within five per cent; poor correlation was obtained using the other materials and techniques.

Experiments performed on the second core were: multiplication runs to criticality; control rod calibration; fuel and boron worth determinations; and three-axes buckling measurements.

Plans for the homogeneous element substitution experiments were advanced. Two types of homogeneous elements are expected to be utilized: one in which all materials in a reference element are homogenized; and a second type in which a single material, (boron or uranium), is homogenized.
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I. INTRODUCTION

This report describes progress during the first quarter on all phases of the ANPP Code Development Program, Pressurized Water Task. The objective of this program is the development of a nuclear physical model and its calculation code for pressurized-water cooled and moderated reactors to predict accurately reactivity, rod positions and power distributions at operating temperatures; and reactor life from a nuclear standpoint. Such a code will be capable of accounting for disadvantage factors of fuel and burnable poison elements, spatial depletion of fuel and burnable poison, spatial distribution of fission products as a function of time, operating history and non-uniform burnup. It will also include such factors as non-uniform fuel loadings and the effects of control rods, boiling and lumped burnable poisons.

A program of critical experiments is being conducted as an integral part of the Code Development Program. The purpose of these experiments is to provide reliable data on which to base the selection of codes and to check the overall accuracy and reliability of the final program.

This report is divided into four main subsections: Theoretical Physics, Calculational Codes, Nuclear Physical Properties and Experimental Physics. Each of these covers a major area of effort in the program. The Theoretical Physics effort involves the investigation and evaluation of various physical models which may be employed. Calculational Codes covers the comparison and, ultimately, the selection of machine programs to be incorporated into the general code. This effort also includes the determination of methods for linking various subroutines and the generation of new machine programs as required. Under Nuclear Physical Properties is included the generation of an accurate, up-to-date, complete file of cross-sections and other pertinent nuclear properties. Finally, Experimental Physics covers the conduct of the experimental program as well as pre-test planning and post-test analysis of results.
II. THEORETICAL PHYSICS*

Three of the physical models used in the analytical methods and techniques described in Ref. 1 have been revised and improved. The affected machine programs are BENIKO, RSP and Programs C and F. The revisions are being completed before automating these and other Ref. 1 techniques. The automation of these programs will provide an efficient calculational capability while more complex and sophisticated models are adopted, developed, programmed and integrated into the desired general code. The resulting pilot code will also provide a test vehicle for rapid checkout of programs which are being evaluated.

The revisions which are being made to the physical models mentioned above are described in the following sections.

A. PROGRAM BRAVE**

The BENIKO buckling and RSP reflector savings physical models (Ref. 1) were combined, modified and extended to provide increased accuracy, convenience, simpler equations and greater flexibility. The revised model, which is utilized in Program BRAVE, incorporates the following features:

1. Point buckling calculation, using a three-point finite difference approximation instead of a four-point approximation.

2. Average group bucklings for each reactor region by flux weighting and volumetric averaging of the point buckling values.

3. Average regional group bucklings by application of the Divergence Theorem to the equations used in (2).

4. Average group bucklings by weighting the point buckling values from (1) over the region of positive thermal bucklings.

5. Reflector savings calculation using the average group buckling values obtained from the methods of (1) through (4) to solve, implicitly, an equation for geometrical buckling as a function of reflector savings.

6. Average the results from methods in (2) or (3) and (5) for each core region to the outer limits of the reactor core.

7. Average group bucklings and reflector savings by means of an iteration technique which varies the reflector savings and also the geometric buckling until the $K_{eff}$ values obtained from the slowing-down and diffusion calculations agree.

* T. M. Olsen
** S. Glasser

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The revisions required to provide the above capabilities are discussed in the following paragraphs. The logical flow of the various physical models is shown in Fig. 1.

Consistent buckling values may be obtained from the leakage term in the few-group, one-dimensional neutron diffusion equation, Eq (1).

\[-D \nabla^2_r \phi_{rj} + \Sigma_j \phi_{rj} = S_j\]  

where

- \( r \) = coordinate direction
- \( \nabla^2_r \) = \( r \) component of Laplacian operator
- \( j \) = energy group index
- \( D \) = diffusion coefficient
- \( \Sigma \) = macroscopic removal cross section
- \( \phi \) = neutron flux
- \( S \) = source term including fission and slowing-down

The term \(-D \nabla^2_r \phi_{rj}\) is the neutron leakage per unit volume in group \( j \) in the \( r \) direction. It may be represented by an equivalent term \(-D_j B_{rj}^2 \phi_{rj}\), in which \( B_{rj}^2 \) is defined as the group \( j \) buckling for the \( r \) direction. This relationship is expressed as Eq (2):

\[D_j \nabla^2_r \phi_{rj} + D_j B_{rj}^2 \phi_{rj} = 0\]  

The equation is not strictly homogeneous, since \( B_{rj}^2 \) is position-dependent. It may be solved for \( B_{rj}^2 \) at any point, \( i \), and an average value obtained by flux-weighted volumetric averaging over the reactor core, Eqs (3) and (4). The group subscript, \( j \), is omitted.

\[B_{ri}^2 = \frac{\nabla^2_r \phi_{ri}}{\phi_{ri}}\]  

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Subroutine Brave

Calculate Point Bucklings for All Reactor Regions if NRMAX > NRMIN or for Core Region if NRMAX = NRMIN or Skip if NOPERA-50 = 3 or 4

NRMAX and NRMIN Define the Regional Limits of the Fissionable Core

NRMAX < NRMIN Suppresses Point Buckling Cals; NRMIN Still Defines Core Reg

NOPERA-50

1 2 3 4

Calculate Average Buckling Between Interfaces for Each Reactor Region

Calculate Average Buckling between Interfaces for Each Reactor Region Using Surface (Gausse) Integral

Calculate Average Buckling and Reflector Savings Using TGST Technique

Core Region Only Over Region of Positive Thermal Bucklings

RSP Calculation for Each Reactor Region Except if NRMAX = NRMIN; Then Do Just for Core Region NRMAX = NRMIN

Average the Average Bucklings over the Core Regions NRMIN Through NRMAX

NRMAX-NRMIN

> 0

Set Average Bucklings and Savings for Non-core Regions Equal to Values for the Single Core Region

Set Perpendicular Average Bucklings and Savings for Perpendicular Flux Calculation Equal to Values Just Calculated for Core Region

Return

Symbols are Defined in Table 1
Variables are Defined in Table 3

Fig. 1. Flow Diagram for Calculating Synthesis Coupling Constants
TABLE 1
Definition of Symbols for
Calculational Flow Diagrams

1. Operation, Function, etc.

Descriptive or working block of
instructions. The box may be
labeled or numbered to assist in
identifying the flow chart with
the code. The box may contain
formulas or substitution expres-
sions. Separate boxes may be used
for input or output operations with
such words as IN, OUT, PRINT, WRITE
or READ in the upper left-hand cor-
ner. Each operation box should have
only one entry point and one exit.

2. Decision or Comparison

Used for conditional or branch opera-
tions. The symbol \( \alpha : \beta \) represents "the
relationship between" \( a \) and \( b \) specified
by the 2 or 3 exits. If the box is drawn
free hand without the use of a template,
an oval or ellipse may be used.

3. Fixed Connector

Used to connect parts of a flow chart.

4. Variable Connector

At a point where control is predetermined
to transfer to one of several possible
points, a variable connector is used. The
setting of variable connectors will be desig-
nated in a separate box, e.g., \( \alpha_1 \leftarrow \alpha_2 \),
\( \alpha \leftarrow \alpha_1 \) or \( \alpha = \alpha_1 \). The arrow when used within
a box is translated "replaces" if it points
to the right and "is replaced by" if it points
to the left. The sign means "is set equal to"
when used in this sense.

5. Closed Subroutine

The notation in the hexagon may refer to
a detailed flow chart contained else-
where or to a standard library subroutine.
Execution of the subroutine occurs at this
point. Multiple exits may also be indicated.

6. Assertion

Various blocks may be attached to the logical
flow lines by dashes or dotted lines to en-
hance the value of the flow chart by indicating:
1. That a counter or instruction is in its
initialized state.
2. To describe a coding trick.
3. To make any parenthetical-type remark
that may serve to clarify the program.

Additional unique symbols for input, output and substitutions could be added but special symbols
are not essential. Use of the fixed connector \( \text{STAR} \) for the point of initiation and \( \text{STOP} \) to
indicate an ending seem desirable, but again special symbols are not essential. Clear designation
of beginning and the ending however is necessary in any flow chart.

Note: A logical rule that should be observed when drawing detailed flow charts is: Every symbol
that appears to the right of an equal sign, at the tail of an arrow, in a variable con-
ector, in a comparison box or in an output box should appear at some previous point in
the flow chart, at the left of an equal sign, at the point of an arrow or in an input box.
The one-dimensional, finite-difference representation of Eqs (3) and (4), using a three-point approximation, is obtained by substitution of the following terms into Eqs (3) and (4), with the integral sign replaced by a summation sign

\[
\frac{1}{B_{ri}} = \frac{\int_{\text{reg}} \nabla^2_{r} \phi_{ri} \, dV}{\int_{\text{reg}} \phi_{ri} \, dV}
\]

(4)

\[
\nabla^2_{ri} \phi_{ri} = \frac{\frac{1}{2} \left( \phi_{i+1} - \phi_{i} \right) - \left( \phi_{i} - \phi_{i-1} \right)}{X_{i+1} - X_{i-1}} \frac{X_{i+1} + X_{i}}{X_{i+1} + 2X_{i} + X_{i-1}} \frac{1}{4} \left( \frac{\phi_{i+1} - \phi_{i}}{X_{i+1} - X_{i}} + \frac{\phi_{i} - \phi_{i-1}}{X_{i} - X_{i-1}} \right)
\]

\[
\phi_{ri} = \frac{1}{4} \left( \phi_{i+1} + 2\phi_{i} + \phi_{i-1} \right)
\]

\[
dV = \left[ \left( \frac{1}{2} \left( X_{i+1} + X_{i} \right) \right)^{G} + \left( \frac{1}{2} \left( X_{i} + 4 \right) \right)^{G} \right] \frac{X_{i+1} - X_{i-1}}{4}
\]

where \( G \) is the geometry constant, with the following values:

0 = slab geometry
1 = cylindrical geometry
2 = spherical geometry

\( dV \) has been replaced by the corresponding geometric volume element using the trapezoidal rule.

\( \phi_{ri} \) is the average value over the finite interval between the dotted lines in the figure below:

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Note that $\nabla^2 \phi_{ri}$ is also the average between these mid-interval dotted lines.

The option for averaging the positive thermal point bucklings is used for only the core region of single-region reactor cores. The average positive bucklings for the remaining energy groups, as well as the average group fluxes, are calculated over the same interval in which the positive thermal bucklings exist. Average group fluxes in each region are calculated for use in determining the reflector savings.

Application of the Divergence Theorem or Gauss (Ref. 1) formula to the flux-weighted average buckling equation, Eq (4), yields a time-saving method of determining average region buckling. The derivation is as follows:

$$\bar{B}_{ri}^{2} = \frac{\int_{\text{reg}} \nabla^2 \phi_{ri} \, dV}{\int_{\text{reg}} \phi_{ri} \, dV} = \frac{\int_{\text{reg}} (\nabla \cdot \nabla \phi_{ri}) \, dV}{\int_{\text{reg}} \phi_{ri} \, dV}$$  \hspace{1cm} (4a)

Gauss' formula: $\int_{S} \vec{F} \cdot d\vec{s} = \int_{V} \nabla \cdot \vec{F} \, dV$  \hspace{1cm} (5)

Replacing the numerator of Eq (4a) by Eq (5), we obtain for the average buckling:

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\[
\overline{B^2_{ri}} = -\int_{\text{reg}} \nabla_r \phi_{ri} \cdot \vec{d}s
\]

where \(\int\) denotes integration over the closed region surface. The one-dimensional finite difference representation of the average buckling is given in Eq (7).

\[
\overline{B^2_{ri}} = -\left(\frac{\phi_{NA} - 1 - \phi_{NA}}{(X_{NA} - 1 - X_{NA})} \left[\frac{1}{2} (X_{NA} - 1 + X_{NA})\right]^G + \frac{\phi_{NZ} + 1 - \phi_{NZ}}{(X_{NZ} + 1 - X_{NZ})} \left[\frac{1}{2} (X_{NZ} + 1 + X_{NZ})\right]^G \right) \left\{\int_{\text{reg}} \phi_{ri} \, dV \right\}
\]

where \(NA\) and \(NZ\) represent the second and next-to-last region points, respectively. The other factors have been previously defined.

If the reactor flux calculation includes more than one core region, the average group bucklings over the core regions, for a particular traverse direction, are obtained by flux-weighting and volumetric averaging of bucklings in each core region, \(K\), using trapezoidal integration. The procedure is expressed mathematically in Eq (8).

\[
\overline{(B^2_r)_{\text{core avg}}} = \frac{\sum_{\text{core}} \overline{B^2_{rK}} \cdot \overline{\phi_{rK}} \cdot V_K}{\sum_{\text{core}} \overline{\phi_{rK}} \cdot V_K}
\]

where:

\[\sum_{\text{core}}\] implies summation over all core regions,

\[V_K = \text{the volume of region } K,\]

\[\overline{\phi_{rK}} = \text{the average flux in region } K.\]

\[
\overline{\phi_{rK}} = \int_{K} \phi_r \, dV_K
\]

The neutron energy index, \(j\), is omitted for clarity.
The reflector savings is determined by solving the expression resulting from equating (for direction r) the total core neutron leakage in the bare core slowing-down calculation, Program C, to that in the reflected core diffusion calculation, Program F.

\[ \sum_{i=1}^{N} (D B_{ri}^2 \phi_i) = \sum_{j=1}^{N} (D B_{rj}^2 \phi_j) \cdot \frac{1}{N} \sum_{j=1}^{N} \phi_j \]  

(10)

The energy group subscripts, i and j, identify the N group terms for the slowing-down and diffusion equations, respectively. The flux ratio term on the right side of Eq (10) is necessary, since the \( \phi_i \)'s are normalized to one source neutron per unit volume in Program C and the \( \phi_j \)'s are normalized to one source neutron in the fissionable core in Program F.

The right side of Eq (10) is evaluated using Eq (4) and the group constants in the slowing-down and diffusion equations. The \( B_{ri}^2 \)'s are the only unknowns in Eq (10) and are obtained from the results of Paragraph 1. They contain the desired reflector savings, \( S_r \), as shown by Eq (11):

\[ B_{ri}^2 = \frac{G^2}{(X_r + 2S_r + 2\delta_i)^2} \]  

(11)

where \( \delta_i \) is the group, i extrapolation distance from Program C, G is \( \pi \), 4.81, or \( \pi \), for slab, cylindrical or spherical geometry, respectively, and \( X_r \) is the core dimension.

The final form of this equivalent leakage method, (ELM), equation is obtained by defining the right side of Eq (10) as \( \alpha \), defining \( D_{ij} G^2 \phi_i / \alpha \) as \( \beta_i \) resulting in Eq (12).*

\[ \sum_{i} \frac{\beta_i}{(X_r + 2S_r + 2\delta_i)^2} = 1 \]  

(12)

*Note in Eq (11) and (12) that \( \delta_i = \delta_j \), since the group constants in the diffusion equation are obtained from the slowing-down calculation. Similarly, \( D_i = D_j \) in Eq (1), (2), and (10).

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For multiregion reactor cores, the regional reflector savings, as determined by the method of Paragraph 4, are averaged over all core regions by means of leakage weighting and are used as the perpendicular value of reflector savings in each core region of a subsequent perpendicular flux calculation. For example, the axial multiregion average reflector savings is used as the perpendicular reflector savings in each radial region in a subsequent radial flux calculation. The average savings over the core is given by Eq (13).

\[
(S_r)_{\text{core reg}} = \frac{\sum_{\text{core reg}} (S_r D_r \overline{B^2_{rj} \cdot \overline{\phi}_{rj}})_K}{\sum_{\text{core reg}} (D_r \overline{B^2_{rj} \cdot \overline{\phi}_{rj}})_K}
\] (13)

where \(D_r \overline{B^2_{rj} \cdot \overline{\phi}_{rj}}_K\) is the neutron leakage per unit volume, in group j, for region K in the r direction.

Another optional method of obtaining consistent transverse and perpendicular bucklings as well as reflector savings for a core region uses the following relationships:

\[
B^2_{Ti} = B^2_{Ai} + B^2_{Ri}
\] (14)

\[
B^2_{Ai} = \left(\frac{\pi}{X_r} + 2\Delta H + 2\delta_i\right)^2
\] (15)

\[
B^2_{Ri} = \left(4.81/X_r + 2\Delta R + 2\delta_i\right)^2
\] (16)

where subscripts, A, R and T refer to axial, radial, and total quantities, respectively.

One assumes a trial value for \(\Delta H\) and obtains \(B^2_{Ai}\) from Eq (15). Then, using the arbitrary value, \(B^2_{Ri}\), of perpendicular buckling which was used in Program F, total buckling is obtained from Eq (14) and used in Eq (17) to solve for \(K_{\text{eff}}\). A new trial value for \(\Delta H\) is used until the value of \(K_{\text{eff}}\) from Eq (17) equals that of Program F. The analogous procedure is then repeated for the perpendicular direction by varying \(\Delta R\) instead of \(\Delta H\).

* F. W. Todt

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Equation (17) is a three-group expression for the reactivity of a bare core.

\[
K_{\text{eff}} = \frac{(1 - P_1)K_1}{(1 + L_1^2 B_{T1}^2)} + \frac{P_1 (1 - P_2)K_2}{(1 + L_1^2 B_{T1}^2)(1 + L_2^2 B_{T2}^2)} + \frac{P_1 P_2 K_3}{(1 + L_1^2 B_{T1}^2)(1 + L_2^2 B_{T2}^2)(1 + L_3^2 B_{T3}^2)}
\]

(17)

where

\[
P_i = \frac{\Sigma_{SL_i}}{\Sigma_{A_i} + \Sigma_{SL_i}}
\]

\[
K_i = \frac{\Sigma_{f_i}}{\Sigma_{A_i}}
\]

\[
L_i^2 = \frac{D_i}{\Sigma_{A_i} + \Sigma_{SL_i}}
\]

where

\[
\Sigma_{SL_i} = \text{slowing-down cross-section}
\]

\[
\Sigma_{A_i} = \text{absorption cross-section}
\]

\[
\nu \Sigma_{f_i} = \text{fission cross-section}
\]

\[
D_i = \text{diffusion coefficient}
\]

The regional slowing-down and diffusion calculations performed in the synthesis require estimates of reflector savings and buckling. Initially, these are not known or known only approximately. The BRAVE physical model allows one to estimate these values and use the approximate results to calculate more accurate values. This iteration process is done according to one of two schemes. The first is for a single region reactor core.
The next slowing-down calculation for each region, for a particular traverse, is given the calculated core values of parallel average-group bucklings or reflector savings. 

The values of parallel average-group bucklings and reflector savings for the core region are used as the perpendicular values in each region of the next perpendicular traverse. For example, when an axial traverse is performed, the calculated axial core bucklings and savings are used as perpendicular values in each radial region when performing a subsequent flux calculation.

The second scheme applies to multiregion reactor cores and is done as follows:

1. The values of parallel average-group bucklings or savings calculated for each region are used in the next slowing-down calculation for each region of a particular traverse.

2. The multiregion average values of average-group buckling, as well as reflector savings given in Eq (17), are used as the perpendicular values in each reactor region for the perpendicular traverse. For example, when an axial traverse is performed, the calculated buckling values for each core region are averaged to give multiregion average values of group bucklings which are then used as the perpendicular average group bucklings for each region of a subsequent radial traverse.

The iteration process is continued until the radial and axial $K_{eff}$ agree. The iteration converges so rapidly that usually one repetition of the original flux and reactivity calculations is sufficient.

The justification for using these methods of specifying parallel and perpendicular values of average-group bucklings and reflector savings is based on previous calculations. The method outlined for single-region reactor cores produces good agreement between the $K_{eff}$ values for each direction used in the synthesis techniques (axial and radial, for example). The method outlined for multiregion reactor cores must be tested after the BRAVE, M and T programs are checked out. Because it is analogous to the single-region core technique, it is expected to produce equally good results.

All revisions to equations, either presented or implied under Section A, were completed during the report periods. Checkout of the revised equations was also completed.
B. PROGRAM M

The GE-ANP Program C, described in Refs. 1 and 2, solves the neutron slowing-down problem for a bare core by Modified Age Theory (an evolved Selengut-Goertzel physical model). The differential equation describing the slowing-down is solved with finite differences using 19 lethargy levels. Heterogeneity, temperature and Behren's corrections are made, if needed.

The original program assumes that the neutron leakage occurring during slowing-down and therefore the neutron energy spectrum, are the same as in a bare core of the same material. For the reactor reflector regions, that is a weak assumption. The assumption is eliminated, at option, in the revision of Program C, renamed M for "Moderation," by substituting the regional leakage calculated by BRAVE.

A second modification was made to correct, at option, for inelastic scattering in stainless steel. The approximate technique described in Ref. 3 and amplified in a private communication from Dr. E. E. Gross of ORNL is used. The inelastic scattering cross-section for stainless steel is approximated by modifying the hydrogen scattering cross-section as follows:

$$\Sigma^i_{SH} (E_j) = \Sigma_{SH} (E_j) + \sum_{i=0}^{j} \xi^i_{in} (E_j) \Sigma_{in} (E_j, E_i)$$

where $\Sigma_{SH}$ is the macroscopic elastic scattering cross-section for hydrogen, $\Sigma_{in} (E_j, E_i)$ is the macroscopic inelastic transfer cross-section for stainless steel for scattering from energy $E_j$ to $E_i$ and

$$\xi^i_{in} (E_j) = \ln (E_j / E_i).$$

This new expression for the hydrogen scattering cross-section replaces the term $\xi \Sigma_{SL}$ on the first line of Ref. 2, page 7.

The revised equations for Program M were written and checked out during the report period.

C. PROGRAM T*

The GE-ANP Program F (Ref. 1) which solves the one dimensional few-group diffusion equation was modified to extend the physical model and equations to solution of the two- or three-group adjoint equations

* D. H. Frederick
for later use in material worth calculations by perturbation theory. The revised program is called T for "Transport." The equations were also modified to provide extra lattice intervals at interfaces and extended to spherical geometry.

If one writes the steady-state diffusion equation as follows:

\[ \nabla \mathbf{J} \phi - K \phi = 0 \]

where \( \phi \) is the flux vector and \( J \) and \( K \) are net gain and net loss matrices respectively, the adjoint flux is defined as the solution of the transposed equation.

\[ \nabla J^+ \phi^+ - K^+ \phi^+ = 0 \]

where \( \phi^+ \) is the adjoint flux and \( J^+ \) and \( K^+ \) are matrix transposes.

Since the input parameters necessary for the solution of the one-dimensional, adjoint, diffusion equations are identical to those necessary for a standard solution and the systems of equations differ only in group-wise term location, it can be seen that the eigenvectors of both systems are, in reality, the same distinct quantity.

The relation of the two systems of equations for an \( n \)-group model is as follows:

Define \( n \) as the maximum group index of the specific model in question.

\[ X_j = 0.0 \quad j > 1 \]
\[ X_j = 1.0 \quad j = 1 \]

where \( X_j \) is the fraction of fission neutrons released in the \( j^{th} \) group.

\[ \nabla J \phi - K \phi = 0 \]

Express the net gain operator of the system as an \( n \times n \) matrix of the form:

MND-C-2200
where $\Sigma_f$ is the fission cross-section.

Express the net loss operator as an $n \times n$ matrix of the form:

$$ J = \begin{bmatrix} \Sigma_{f_1}^{(n)} & \Sigma_{f_2}^{(n)} & \ldots & \Sigma_{f_n}^{(n)} \\ 0 & 0 & \ldots & 0 \\ \vdots & \vdots & \ddots & \vdots \\ 0 & 0 & \ldots & 0 \end{bmatrix} $$

where $D$ is the diffusion coefficient, $\Sigma_r$ the total absorption cross-section and $\Sigma_{SD}$ the slowing-down cross-section.

Upon expansion of the above parameters for a three-group model, the following equations are obtained:

$$ \vec{\nabla} \cdot D_1 \vec{\nabla} \phi_1 - \Sigma_{r_1} \phi_1 + \nu \Sigma_{f_1} \phi_1 + \nu \Sigma_{f_2} \phi_2 + \nu \Sigma_{f_3} \phi_3 = 0 $$

$$ \vec{\nabla} \cdot D_2 \vec{\nabla} \phi_2 - \Sigma_{r_2} \phi_2 + \Sigma_{SD_1} \phi_1 = 0 $$

$$ \vec{\nabla} \cdot D_3 \vec{\nabla} \phi_3 - \Sigma_{r_3} \phi_3 + \Sigma_{SD_2} \phi_2 = 0 $$

The operator transposes are:

$$ J^+ = \begin{bmatrix} \Sigma_{f_1} & 0 & 0 \\ \Sigma_{f_2} & 0 & 0 \\ \Sigma_{f_3} & 0 & 0 \end{bmatrix} $$

MND-C-2200
where $J^+$ and $K^+$ are transposes of the matrices $J$ and $K$, respectively.

Adjoint fluxes, $\phi^+$, are defined as above by the following equation:

$$\nabla J^+ \phi^+ - K^+ \phi^+ = 0$$

Upon expansion, the resultant equations are:

$$\nabla \cdot D_1 \phi_1^+ - \Sigma_{r_1} \phi_1^+ + \Sigma_{SD_1} \phi_2^+ + \nu \Sigma_t \phi_1^+ = 0$$

$$\nabla \cdot D_2 \phi_2^+ - \Sigma_{r_2} \phi_2^+ + \Sigma_{SD_2} \phi_3^+ + \nu \Sigma_t \phi_1^+ = 0$$

$$\nabla \cdot D_3 \phi_3^+ - \Sigma_{r_3} \phi_3^+ + \nu \Sigma_t \phi_1^+ = 0$$

In the adjoint solution, the energy-dependent source terms take the following form:

$$S_1 = \Sigma_{SD_1} \phi_2^+ + \nu \Sigma_t \phi_1^+$$

$$S_2 = \Sigma_{SD_2} \phi_3^+ + \nu \Sigma_t \phi_1^+$$

$$S_3 = \nu \Sigma_t \phi_1^+$$

Program T uses these source terms if adjoint fluxes are requested.
To permit calculation of regional buckling by the Gauss Method, it was necessary to have an extremely small lattice point spacing at boundaries and regional interfaces so that minor changes in the first derivative of the flux could be accurately calculated. This was accomplished by insertion of a system of equations which enters extra lattice points at the above locations. The spatial increment for these points, Delta, is an input variable. The following diagrams illustrate the procedure:

Given a three-region problem,

<table>
<thead>
<tr>
<th>Region</th>
<th>Interval Width (cm)</th>
<th>Number of Intervals</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1</td>
<td>6</td>
</tr>
<tr>
<td>2</td>
<td>1/2</td>
<td>7</td>
</tr>
<tr>
<td>3</td>
<td>3/4</td>
<td>3</td>
</tr>
</tbody>
</table>

Basic Mesh (prior to insertion of extra points)

The F-3 (Ref. 4) calculation was revised to include computations for spherical geometry. The previous equations for slab and cylinder were generalized so that a geometry factor, G, could be used. Generalized, energy-independent, geometric weighting factors, Beta, Delta and Gamma were obtained from the two-energy group diffusion calculation, F-2 (Ref. 5) and are listed below.
\[ \beta_{iP} = \frac{r_{iP + 1} - r_{iP}}{r_{iP} - r_{iP-1}} \cdot \frac{1 + \frac{r_{iP} + r_{iP - 1}}{3r_{iP} - r_{iP} - 1}}{1 + \frac{r_{iP + 1} + r_{iP}}{3r_{iP} - r_{iP} + 1}} G \]

\[ \gamma_{iP} = \frac{r_{iP + 1} - r_{iP}}{2 (G + 1)} \cdot \frac{(r_{iP} + r_{iP + 1}) - (3r_{iP} - r_{iP} + 1)}{r_{iP} + r_{iP + 1}} G \]

\[ \delta_{iP} = \frac{r_{iP + 1} - r_{iP}}{2 (G + 1)} \cdot \frac{(3r_{iP} - r_{iP-1}) - (r_{iP} + r_{iP-1})}{3r_{iP} - r_{iP} - 1} G \]

where \( iP \) is the lattice point index, \( iP = 1, 2, 3, \ldots \)

\[ r \] = the lattice point position

\[ G \] = the geometry factor: 0 for slab, 1 for cylinder and 2 for sphere.

The revised equations for Program T were written and checked out during the report period.
III. CALCULATIONAL CODES*

The major calculations which must be performed by Army Pressurized Water Reactor Code (APWRC) are indicated in the calculational flow diagrams shown in Figs. 2 through 7**. These flow diagrams were first presented in Ref. 6 and are repeated here for convenience. One of the major tasks in the development of the general code is the selection of specific programs to perform the many calculations indicated in the flow diagrams.

A. STATUS OF VARIOUS PROGRAMS

In selecting a machine program for a particular application, it has been decided to favor programs coded in FORTRAN. The type of source program language (FORTRAN or machine language) is considered of greater importance to the APWRC Development Program than accuracy, running time or theory, since these do not vary greatly from program to program and can be modified more readily than the type of source program language. Therefore, in any case where the equations describing a particular physical model have been programmed in both FORTRAN and machine language, the FORTRAN version will be selected. If several FORTRAN versions exist, a comparison of accuracy, running time and theory will be necessary. If no FORTRAN version exists, the available machine language versions will be compared for accuracy, running time and theory.

To avoid comparison of programs which may become obsolete before their application is necessary, comparisons will be made as the need for application arises. To date, no comparisons have been made because not all pertinent programs have been received.

Table 2 summarizes the status of each machine program of interest at the end of the quarter. For each program, the theory, number of dimensions, primary function of the code, etc., are listed. The numbers in parentheses following the function of each program indicate the figure numbers of the flow diagrams in which that particular code is likely to be used. Where a comparison between two codes remains to be made, both codes are shown in Table 2.

All of the codes in Table 2 are available or have been ordered. A large number of other codes which will be of interest in the near future have also been ordered. However, some installations have not responded

* T. M. Olsen
** Symbols used in these figures are defined in Table 1.
Fig. 2. Flow Diagram for Reactivity, Critical Mass, and Critical Size
Fig. 3. Flow Diagram for Rod Positions Versus Time and Reactivity Versus Time During Reactor Core Burnup
When performing the calculation for

\[
\begin{bmatrix}
\text{cell corrections} \\
\text{rod heterogeneity corrections} \\
\text{lumped poison heterogeneity corrections} \\
\text{buckling and savings} \\
\text{void distributions}
\end{bmatrix}
\]

the calculation will be performed every

\[
\begin{bmatrix}
\Delta NFCC \\
\Delta NRHC \\
\Delta NLBP \\
\Delta NBRS \\
\Delta NVD
\end{bmatrix}
\]

iterations (time intervals). Each time the

\[
\begin{bmatrix}
\text{Cell correction} \\
\text{Rod heterogeneity correction} \\
\text{lumped poison heterogeneity correction} \\
\text{buckling and saving} \\
\text{Void distribution}
\end{bmatrix}
\]

calculation of

\[
\begin{bmatrix}
\text{NFCC} \\
\text{NRHC} \\
\text{NLBP} \\
\text{NBRS} \\
\text{NVD}
\end{bmatrix}
\]

is increased by

\[
\begin{bmatrix}
\Delta NFCC \\
\Delta NRHC \\
\Delta NLBP \\
\Delta NBRS \\
\Delta NVD
\end{bmatrix}
\]

performed the counter

\[
\begin{bmatrix}
\text{NFCC} \\
\text{NRHC} \\
\text{NLBP} \\
\text{NBRS} \\
\text{NVD}
\end{bmatrix}
\]

The count on the number of iterations (\(N_I\)) is then

\[
\begin{bmatrix}
\text{NFCC} \\
\text{NRHC} \\
\text{NLBP} \\
\text{NBRS} \\
\text{NVD}
\end{bmatrix}
\]

checked against

and the decision made

\[
\begin{bmatrix}
\text{Cell corrections} \\
\text{Rod heterogeneity corrections} \\
\text{lumped poison heterogeneity corrections} \\
\text{buckling and savings} \\
\text{void distributions}
\end{bmatrix}
\]

to calculate or not calculate the

\[
\begin{bmatrix}
\text{NFCC} \\
\text{NRHC} \\
\text{NLBP} \\
\text{NBRS} \\
\text{NVD}
\end{bmatrix}
\] =

\[
\begin{bmatrix}
\Delta NFCC \\
\Delta NRHC \\
\Delta NLBP \\
\Delta NBRS \\
\Delta NVD
\end{bmatrix}
\]

 Initially
NREPET is set internally by the program. Its function is to allow the program to backtrack

\[
\begin{bmatrix}
\text{cell corrections} \\
\text{rod heterogeneity corrections} \\
\text{lumped-poison heterogeneity corrections} \\
\text{every bucklings and savings} \\
\text{void distributions}
\end{bmatrix}
\]

time a new void distribution is calculated

\[
\begin{bmatrix}
\Delta\text{NFCC} \\
\Delta\text{NRHC} \\
\Delta\text{NLBP} \\
\Delta\text{NBR} \\
\Delta\text{NVD}
\end{bmatrix}
\]

and every \(\frac{\text{iteration}.}{\text{iteration}}\)

\[
\begin{bmatrix}
\Delta\text{NFCC} \\
\Delta\text{NRHC} \\
\Delta\text{NLBP} \\
\Delta\text{NBR} \\
\Delta\text{NVD}
\end{bmatrix}
\]

The are input parameters.
Any Number of Groups May Be Used, Including Epithermal Groups and Subgroups Within the Thermal Group. The latter, if used, will account for Thermal Spectrum Hardening in the Fuel Regions

Determine Macroscopic Cross Sections for Each Cell Reg.

Select Type of Flux Calculation

Determine the Two-Dimensional, Few-Group Flux Distribution Using a \( P_n \) or \( S_n \) Solution of the Boltzmann Transport Eq.

Determine the Two-Dimensional Few-Group Flux Distribution Using a Monte Carlo Solution of the Boltzmann Transport Eq.

Determine the Two-Dimensional Heterogeneity Corrections for Each Group and for Each Material from the Calculated Flux Distribution

TOYOKO Determines if Another Problem of the Same Type is to be Run, or a New Problem of Another Type, or if All Calculations are Finished

Summarize and Print Results Return to TOYOKO

Monte Carlo

Only One of the Two Options, \( P_n \) or \( S_n \), Will Be Available. The Choice Will Depend on the Available Programmed Solutions

The Tubular Fuel Cell Cannot Be Analyzed in Two Dimensions in a Plane Perpendicular to Its Axis Except in RG Geometry Since the Center of the Coordinate System and the Cell Axis Must Coincide. RG Geometry is not of Interest Since Fuel Tubes are Loaded with \( G \) Symmetry

Fig. 4. Flow Diagram for Unsynthesized Relaxed Two-Dimensional Fluxes in Flat Plate Type Fuel Cell*

MND-C-2200
Read and Print All Input Data and Check for Errors

1

Determine Regional Multi-group Fuel Region Heterogeneity Corrections; 2D for Plates, 1D for Tubes

Determine Slowing-Down Spectrum and Few-Group Constants for Each Reactor Region Including Control Rod and Lumped Burnable-Poison Regions, If Present

Select Type of Flux Calculation

Three-Dimensional Few-Group Multiregion Diffusion Theory

Three-Dimensional Few-Group Multiregion Monte Carlo Theory

Three Dimensional Few-Group Multiregion Transport Theory

2

Is Reactor in Boiling Phase?

Yes

Determine Three-Dimensional Temperature Distribution and Correct Regional Moderator Volume Fractions for Resulting Void Formation

3

Is $k_{eff} - k'_{eff} < D_{max}$?

Yes

or is $N_I > N_I_{max}$?

No

Set $N_I = N_I + 1$

4

Prime Denotes Results of Previous Iteration. $k_{eff}$ is Initially Zero. $D_{max}$ is Convergence Criterion. $N_I$ is the Iteration Count. $N_I_{max}$ is the Maximum Permissible Number of Iterations. $D_{max}$ and $N_{max}$ are Input Parameters.

TOYOKO Determines if Another Problem of the Same Type is to be Run or a New Problem of Another Type or If All Calculations are Finished

Summarize and Print Results Return to TOYOKO

Fig. 5. Flow Diagram for Unsynthesized Relaxed Three-Dimensional Gross Flux Distribution

MND-C-2200
Read and Print All Input Data and Check for Errors

- Determine Necessary Core Characteristics: Neutron Lifetime, Void Coefficient, Temperature Coefficient and Effective Delay Fraction.
  - Set $k_{eff} = 1$

- Determine New Value of Reactivity, After a Time Interval $\Delta t$, Due to an Arbitrary Reactivity Input from Rods or Water Level, Including Filling from Water Storage Tank.

- Determine the New Values of Flux and Power After the Time Interval $\Delta t$.

- Determine the Change in Temperature and Temperature Distribution.

- Select Fuel Element Geometry for Calculating Void Distribution.
  - Laminated Fuel Bundle
  - Ceramic Flat Fuel Plate
  - Ceramic Tubular Fuel Element

- TOYOKO Determines if Another Problem of the Same Type is to be Solved, or a New Problem of a Different Type, or if All Calculations are Finished.
  - Summarise and Print Results Including Peak Power and Temperature, Energy Release, etc.; Return to TOYOKO.

- The Reactor is Assumed to be Stable if $k_{eff} \leq 1$ for $N$ Iterations, a Time $\Delta t$. If Input Data or May be Computed by the Program.

- NIMAX is Given as Input

- Is $NI = N_{stbl} + NI$?
  - Yes
  - Set $N_{stbl} = NI$
  - Is $NI > N_{stbl}$?
    - Yes
      - Correct Reactivity for Effects of Temperature
      - Correct Reactivity for Effects of Void
      - Determine the Change in Void Content and Distribution Due to Steam and Radiolytic Gas Production
    - No
      - Set $N_{stbl} = NI - 1$
  - No

- Is $NI \leq N_{stbl}$?
  - Yes
    - Correct Reactivity for Effects of Temperature
    - Correct Reactivity for Effects of Void
  - No
    - Set $NI = NI + 1$

- Is $NI > N_{stbl}$?
  - Yes
    - Set $NI = NI + 1$
  - No

Fig. 6. Flow Diagram for Transient Properties

MND-C-2200
Start

Read and Print the Dependent Parameter and the N Independent Parameters, and Their Ranges, Which are to be Investigated. Check for Errors

Select and Calculate Design Points for Fractional Replication of the Composite Design Representing the N Parameters and Their Ranges

Transfer to the Appropriate Section of the Code ("Reactivity, Critical Size...", "Rod Position versus Time...", "Transient...") Containing the Dependent and N Independent Parameters

Return After Determining the Value of the Dependent Parameter at Each Design Point

Determine the Coefficients in an N-Dimensional Quadratic Fit to the Dependent Parameter at Each Design Point

Determine the Goodness-of-Fit, Residual, Standard Deviation, and Coefficient of Variation for the Approximation

Summarize and Print Results and Return to TOYOKO

TOYOKO Determines if Another Problem of the Same Type is to be Solved, or a New Problem of a Different Type, or if All Calculations are Finished

Punch Cards Off-Line for Diametric Projection on Benson-Lehrer Plotter, of the N-Dimensional Dependent Variable Surface

See, e.g., MND-E-1744, Fig. 7

Fig. 7. Flow Diagram for Parametric Study Using Synthetic Design Techniques
<table>
<thead>
<tr>
<th>Program</th>
<th>Theory</th>
<th>Number of Dimensions</th>
<th>Function</th>
<th>Source Language</th>
<th>Status</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>M</td>
<td>Modified age</td>
<td></td>
<td>Generation of group constants (2, 3, 5)</td>
<td>FORTRAN</td>
<td>Being checked out</td>
<td>See Section II-2</td>
</tr>
<tr>
<td>T</td>
<td>Transport (P1)</td>
<td>One</td>
<td>Calculation of $K_{\text{eff}}$ (2,3)</td>
<td>FORTRAN</td>
<td>Being checked out</td>
<td>See Section II-3</td>
</tr>
<tr>
<td>BRAVE</td>
<td></td>
<td></td>
<td>Generation of coupling constants (2,3)</td>
<td>FORTRAN</td>
<td>Being checked out</td>
<td>See Section II-1</td>
</tr>
<tr>
<td>REIKO</td>
<td>Successive collision; slab geometry</td>
<td>One</td>
<td>Calculates cell corrections (2, 3, 5)</td>
<td>FORTRAN</td>
<td>Available</td>
<td>Described in Ref. 1</td>
</tr>
<tr>
<td>TCP</td>
<td>Successive collision; cylindrical geometry</td>
<td>One</td>
<td>Calculates cell corrections (2, 3, 5)</td>
<td>FORTRAN</td>
<td>Available</td>
<td></td>
</tr>
<tr>
<td>1050</td>
<td>Successive collision; cylindrical geometry</td>
<td>One</td>
<td>Calculates transmission coefficients (2, 3)</td>
<td>FORTRAN</td>
<td>Report on order from GE-ANP</td>
<td>Will be compared with TCP for running time</td>
</tr>
<tr>
<td>AMD-112</td>
<td>Transport ($S_n$)</td>
<td>One</td>
<td>Fine and gross flux distribution; cell corrections (2, 3, 5)</td>
<td>FORTRAN</td>
<td>On order from ANL</td>
<td>Will be compared with Martin SN</td>
</tr>
<tr>
<td>Martin SN</td>
<td>Transport ($S_n$)</td>
<td>One</td>
<td>Fine and gross flux distribution; cell corrections (2, 3, 5)</td>
<td>FORTRAN</td>
<td>Available</td>
<td></td>
</tr>
<tr>
<td>SN1 CYL CELL N REGIONS (C)</td>
<td>Transport ($S_n$)</td>
<td>One</td>
<td>Calculates cell corrections (2, 3, 5)</td>
<td>FORTRAN</td>
<td>Available</td>
<td></td>
</tr>
<tr>
<td>REM</td>
<td>Transport (P1); few group</td>
<td>Two</td>
<td>Calculates $K_{\text{eff}}$ and 2D flux distribution (2, 3)</td>
<td>704 SAP</td>
<td>On order from KAPL through ARM</td>
<td></td>
</tr>
<tr>
<td>TKO</td>
<td>Transport (P1); few group</td>
<td>Three</td>
<td>Calculates $K_{\text{eff}}$ and 3D flux distribution (5)</td>
<td>704 SAP</td>
<td>On order from W-BAPD through ARM</td>
<td></td>
</tr>
<tr>
<td>TURBO</td>
<td>Transport (P1); few group</td>
<td>Two</td>
<td>Calculates $K_{\text{eff}}$ versus time (3)</td>
<td>704 SAP</td>
<td>Available</td>
<td></td>
</tr>
<tr>
<td>DRACO-5</td>
<td>Transport (P1); few group</td>
<td>Three</td>
<td>Calculates $K_{\text{eff}}$ versus time (3)</td>
<td>704 SAP</td>
<td>On order from W-BAPD through ARM</td>
<td></td>
</tr>
<tr>
<td>ABRAC-1</td>
<td>Transport (P1); few group</td>
<td>Three</td>
<td>Calculated $K_{\text{eff}}$ versus time (3)</td>
<td>704 SAP</td>
<td>On order from W-BAPD through ARM</td>
<td>Version of DRACO including local boiling heat transfer and fluid flow equations.</td>
</tr>
<tr>
<td>SYD</td>
<td>Fractional replication of composite design</td>
<td></td>
<td>Parametric study of system variables (7)</td>
<td>FORTRAN</td>
<td>Available</td>
<td></td>
</tr>
</tbody>
</table>
to the request for a particular program, replied that their schedules did not provide for the dissemination of such information or expressed proprietary rights. For programs which have been developed under government contract, requests will be directed to the authors through the Atomic Energy Commission.

B. STATUS OF PILOT CODE

The pilot code mentioned briefly in Chapter II was begun by automating the Program M, Program T and BRAVE computations. The resulting linkage has been designated SYNFAR for "Synthesized Flux and Reactivity." The primary computation performed by this linkage is core reactivity; synthesis of the two-dimensional flux distribution is used. The linkage is the foundation of the flow charts for reactivity, critical size, critical mass and reactor lifetime shown in Figs. 2 and 3.

The logical flow used to automate the calculations is shown in Fig. 8. The flow chart symbols are defined in Table 1. Control words and subroutine functions are defined in Table 3.

The program is designed for flexibility. The subroutines BRAVE, M and T are logical extensions of Programs BENIKO, C and F. That is, the latter three programs initially will be used, respectively, to provide synthesis coupling constants to solve the moderation problem by determining neutron flux-versus-lethargy and flux-weighted few-group constants, and to solve the transport problem and determine few-group neutron flux-versus-position by the diffusion theory approximation. Then, if desired at some later date, a more sophisticated physical model and equations may be substituted for any or all of the three subroutines. An SN approximation may be substituted for the diffusion theory approximation, for example, to solve the transport problem.

The program can be used initially in binary object deck form. It is also designed for use with the library tape system described below. A third option is application in APWRC. In this latter case, the program will also be used with the library tape system which will combine it automatically with other calculations--cell corrections, for example--to completely automate the Ref. 1 methods and techniques and also to perform the computations shown in the flow diagrams in Figs. 2 through 7.

Regardless of the mode of application, the program conforms to the data-checking requirement dictated in Ref. 6. All input data is loaded into the machine and checked before any computations are performed.
TABLE 3
Definition of Quantities in Figs. 1 and 8

Control Words and Variables and Their Functions

<table>
<thead>
<tr>
<th>Word</th>
<th>Value</th>
<th>Function</th>
</tr>
</thead>
<tbody>
<tr>
<td>NOPERA</td>
<td>1-25</td>
<td>Region index for moderation calculation</td>
</tr>
<tr>
<td></td>
<td>&gt; 49</td>
<td>Few-group transport calculation</td>
</tr>
<tr>
<td>NRECRD</td>
<td>0</td>
<td>There are more cases to follow</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Last case</td>
</tr>
<tr>
<td>NCHAN</td>
<td>0</td>
<td>There are no change cards to be read</td>
</tr>
<tr>
<td></td>
<td>&gt; 0</td>
<td>This case is similar to the previous one except for the changes on change cards (NCHAN is the number of variables to be changed)</td>
</tr>
<tr>
<td>NBRAVE</td>
<td></td>
<td>Flow control word initially zero and reset according to the flow diagram</td>
</tr>
<tr>
<td>NRSTRT</td>
<td></td>
<td>Flow control word initially zero and reset according to the flow diagram</td>
</tr>
<tr>
<td>NSTART</td>
<td>0</td>
<td>Standard flat source guess</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Initial source distribution is that of previous converged case</td>
</tr>
<tr>
<td>N27</td>
<td></td>
<td>Geometry control word for few-group transport calculation</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Slab</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Cylinder</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>Sphere</td>
</tr>
<tr>
<td>EFFKM</td>
<td></td>
<td>$K_{\text{eff}}$ from neutron moderation calculation</td>
</tr>
<tr>
<td>EFFK1</td>
<td></td>
<td>$K_{\text{eff}}$ from few-group transport calculation, slab geometry</td>
</tr>
<tr>
<td>EFFK2</td>
<td></td>
<td>$K_{\text{eff}}$ from few-group transport calculations, cylindrical geometry</td>
</tr>
<tr>
<td>NRMIN</td>
<td></td>
<td>Minimum fissionable region number</td>
</tr>
<tr>
<td>NRMAX</td>
<td></td>
<td>Maximum fissionable region number</td>
</tr>
</tbody>
</table>

Subroutines and their Functions

1. TPSET
Allows the user to select the tape unit numbers to agree with the convention at a specific installation. At present, these are initialized to the IBM-704 MONITOR convention in MND-E-1718-2.

2. CHANGE
Permits data changes to be read into storage in order to eliminate duplication of input data for a case which is similar to one already in storage.

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<table>
<thead>
<tr>
<th></th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td><strong>TINPUT</strong>&lt;br&gt;Reads the input data for the few-group transport calculations from input tape or cards and checks for errors.</td>
</tr>
<tr>
<td>4</td>
<td><strong>TWTBIN</strong>&lt;br&gt;Writes the few-group transport calculation input data on a binary tape for intermediate storage.</td>
</tr>
<tr>
<td>5</td>
<td><strong>MINPUT</strong>&lt;br&gt;Reads the input data for the neutron moderation calculation from input data or cards and checks for errors.</td>
</tr>
<tr>
<td>6</td>
<td><strong>MWTBIN</strong>&lt;br&gt;Writes the neutron moderation calculation input data on a binary tape for intermediate storage.</td>
</tr>
<tr>
<td>7</td>
<td><strong>BSFILE</strong>&lt;br&gt;Backspaces the intermediate data storage tape to the beginning of the file when input data for all cases has been written on the tape, or when another BRAVE coupling coefficient iteration is done.</td>
</tr>
<tr>
<td>8</td>
<td><strong>TRTBIN</strong>&lt;br&gt;Reads the input data for the few-group transport calculation from the intermediate binary data storage tape.</td>
</tr>
<tr>
<td>9</td>
<td><strong>T</strong>&lt;br&gt;Solves the one-dimensional transport equation</td>
</tr>
<tr>
<td>10</td>
<td><strong>TOUTPU</strong>&lt;br&gt;Writes the output from the &quot;T&quot; calculation</td>
</tr>
<tr>
<td>11</td>
<td><strong>MRTBIN</strong>&lt;br&gt;Reads the input data for the neutron moderation calculation from intermediate binary storage tape.</td>
</tr>
<tr>
<td>12</td>
<td><strong>M</strong>&lt;br&gt;Solves the neutron moderation equations</td>
</tr>
<tr>
<td>13</td>
<td><strong>MOUTPU</strong>&lt;br&gt;Writes the output from the &quot;M&quot; calculation.</td>
</tr>
<tr>
<td>14</td>
<td><strong>BRAVE</strong>&lt;br&gt;Buckling, reflector savings and flux averaging calculation of synthesis coupling coefficients.</td>
</tr>
</tbody>
</table>
TABLE 3 (continued)

15. **BRAVEO**  
   Writes the output from the "BRAVE" calculation.

16. **NFLT**  
   See Chapter III, Section C.
The details of each of the subroutines in Fig. 8 were resolved and described in flow charts. The main program and subroutines will be assembled and checked out during the next quarter, and the next quarterly report will include a detailed description of the program and its application.

C. REVISED FORTRAN LIBRARY TAPE SYSTEM

The Nuclear FORTRAN Library Tape System, developed previously and described in Ref. 1, was revised and almost fully checked out. The original version was written to permit convenient recording of many machine programs on a magnetic tape for later use as desired. The old system had a major disadvantage of requiring excessive search time for programs recorded far from the beginning of the tape. Another disadvantage was the restriction that each program written on the tape contain an error diagnostic routine. These disadvantages have been eliminated in the new system. In addition, several features, described in the last paragraph below, were added to automate the various programs required by APWRC.

The revised and added features are:

1. The new system will search for the next record to be used while computations are being performed under the current record.

2. The tape will not rewind between MID card changes. This feature, together with the feature of zero or minimum search time will make the tape much more efficient than the old version.

3. A new "700" card is necessary with the new system. The new card contains 70X 70Y in the first seven columns on the card and must precede each data case. The decimal number of the record desired is X and Y is the decimal number of the next record to be used. Record Y will be searched for during computations under record X.

4. A return to MONITOR, the Martin Operator System (Ref. 1), will result on the 704 (or to CHAOS on the 709) if record number 0 is called. There is no "704" card as with the old system. Instead, a special card "702 702" is substituted by the machine operator for the last card, "700 700," in a data deck for the library system.
Fig. 8. SYNFA Program Logical Flow Diagram
The only restrictions necessary in order to write a program on the system tape are the following two:

1. The first statement in the program must be CALL NFLT
2. The last statement in the program must be a PAUSE or STOP.

The subroutine NFLT may be placed anywhere in the compiled object deck. Its function is to remove, before computations under the main program, any HPR instructions in the object deck and replace them with TSX's to a diagnostic routine in the subroutine so that instead of stopping on any HPR instruction, the program will print out the following diagnostic:

*HPR 00000,0 EXIT FROM RECORD 00 ON CARD 00 FROM REAR OF PROBLEM*

The zeros, of course, are variable.

This diagnostic will be printed out if the program is being used from the tape. If the program is being used from the deck, the diagnostic feature will still work; the words "FROM PROGRAM" will be printed instead of "FROM RECORD 00."

At this point, the subroutine NFLT will have skipped over the remaining data cards in the problem and found the next "70X 70Y" card. If X equals zero, the subroutine NFLT will transfer control back to MONITOR on the 704 or CHAOS on the 709. If X is non-zero and different than it was for the previous data case, then record X will be loaded into the machine. If the program was run from the deck and X is non-zero and different than it was for the previous data case, then the next program deck will be loaded into the machine. If the programs are being loaded from their deck versions, the "Y" on the "700" card will be ignored. Under operation from the tape version, record Y will be searched for during computations under record X.

If any information is written on a magnetic tape for later punching off-line, the standard operating procedure under MONITOR or CHAOS is for the on-line printer to notify the operator to "punch out" the tape and then for the machine to stop and proceed after the operator reads the message, takes the necessary action and pushes the START button. These machine functions will be done if any data case is followed by a "700" card with X equal to 99. Under tape operation, the signal to the operator will be given at the end of the library run. Under deck operation, the signal will be given when the "799 70Y" card is read in. In both cases, Y is ignored.
The computations performed by various APWRC records on the library tape will be combined and sequenced automatically as follows: There will be a Master Control Record on the tape for each flow chart shown in Figs. 2 through 7. A particular Master Control Record will be selected with the appropriate "70X 70Y" card as explained above. This record will then read in a data card containing the number of each record to be used in the computation described by the corresponding flow chart and also the frequency of application of each record. For example, in a reactor burnup calculation, the record calculating lumped burnable poison corrections may be desired only for every fifth time step. The record and frequency numbers will be stored in memory by the Master Control Record and used later, as needed by the system loader.

D. MISCELLANEOUS CHECKED-OUT SUBROUTINES*

Two subroutines required by the SYNFAR flow chart (Fig. 8), but generally applicable, were written and checked out.

Subroutine CHANGE. This subroutine will read new values of any desired variables in either fixed or floating point form and store these new values over the previous ones.

In order to use this subroutine, the following data must be given:

1) Number of changes (NCHAN), defined as the number of non-subscripted variables plus the number of subscripted variable arrays, if any, to be changed.

2) The absolute address (NADRSS) of the variable to be changed (from the FORTRAN assembly map and loader increment).

3) Length of the array (LENGTH) if a subscripted variable is to be changed (zero if nonsubscripted).

4) Type of variable (NTYPE), either fixed or floating point, denoted, respectively, by a 1. or 2.

The subroutine has been written so that the control word (NCHAN) in the program calling CHANGE is also the number of variables to be changed, thereby eliminating the need for another control word in the subroutine.

* D. E. Wolf
The subroutine also has a number of input data error checks in the form of PAUSE statements. If an error is encountered while reading input data, the machine will halt with a characteristic number appearing in the address field of the storage register on the console. By comparing this number with those given in the table at the beginning of the CHANGE source deck listing, the error can be determined.

The calling statement for this subroutine is:

**CALL CHANGE (NT9, NCHAN).**

Below is a sample preparation of input data. Consider the array \( A_1, A_2, A_3 \) and \( LA \) as the variables to be changed. From the storage map and loader increment, the variable \( A \) has a decimal location of 243; the array to be changed has a length of 3; and the variable \( A \) is in floating point form. Therefore:

\[
\text{NADRSS} = 243, \text{LENGTH} = 3 \text{ and } \text{NTYPE} = 2.
\]

Assume that \( LA \) has a decimal location of 560; the variable is non-subscripted, so it has no array; and the variable is in fixed-point form. Therefore:

\[
\text{NADRSS} = 560, \text{LENGTH} = 0 \text{ and } \text{NTYPE} = 1.
\]

Note that \( NCHAN \) the number of variables changed in this case is 2. The change cards would be arranged as follows:

**CARD 1** 243 3 2  
**CARD 2** THREE NEW VALUES OF A (I)  
**CARD 3** 560 0 1  
**CARD 4** ONE NEW VALUE OF LA.

**Subroutine TPSET.** - If the APWRC programs contained fixed IBM-704 logical tape unit numbers in the FORTRAN output statements, these programs could not be conveniently used at another installation having a different tape convention. In that case, in order to change the convention every FORTRAN output statement would have to be revised and the entire program reassembled. To avoid this problem, variable tape unit numbers were used in the source programs. However, this requires the variable tape numbers to be initialized to the desired tape numbers. Therefore, a short FORTRAN subroutine was written and used to initialize the logical tape unit numbers to those consistent with MONITOR. If it is ever necessary to change this convention, the TPSET source program can be easily modified and reassembled with a minimum expenditure of machine time.

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Note that TPSET is unnecessary with IBM-709 FORTRAN, since this function is handled by 709 FORTRAN subroutine UNIT.

As it is now written, subroutine TPSET is suitable for use with any program using variable logical tape unit numbers under the MONITOR or CHAOS convention. The calling statement for this subroutine is:

```
CALL TPSET (NT1, NT2, NT3, NT4, NT5, NT6, NT7, NT8, NT9, NT0).
```

An example of the use of TPSET is shown in Fig. 8.
The development of a consistent set of cross-sections and other nuclear properties from up-to-date basic data is necessary in order to maintain a code that is reliable and flexible.

A program was drawn up to obtain this basic cross-section data from various sources. Work that has been done by other installations will be used after first checking the data for accuracy. The reports BNL-325, second edition, and UCRL-5226 will be relied upon heavily until more recent information becomes available. The Breit-Wigner analysis will be used in the resonance region, whenever possible, to complement the insufficient amount of measured data. The energy dependence of the parameter $\xi$, the average logarithmic energy decrement per collision, and $\bar{\mu}_L$, the average cosine of the scattering angle due to anisotropic scattering will be included. An inelastic scattering matrix, employing appropriate theoretical models when necessary, will be developed.

The types of nuclear properties to be obtained for use in APWRC are listed:

1. Absorption cross-section* - $\sigma_a$
2. Fission cross-section* - $\sigma_f$
3. Neutrons released per fission - $v$
4. Elastic scattering cross-section* - $\sigma_{el}$
5. Average logarithmic energy decrement - $\xi$
6. Inelastic scattering cross-section - $\sigma_{in}$
7. Fission spectrum - $F$
8. Average cosine of the scattering angle - $\bar{\mu}_L$

The cross-sections with an asterisk will have, in addition, 16 thermal values, computed by averaging the cross-section over a Maxwell-Boltzmann distribution. 1001 energy points will be represented.

* R. Hubner, M. J. Kniedler, J. B. Weddell

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A. COMPILATION OF BASIC DATA

Compilation of the basic cross-section data is being achieved by the mechanical reading of curves by a Benson-Lehner machine which produces data punched on cards. These cards are acceptable for use as input for the program described in Chapter III.

Cross-sections are being read in the energy range 0.001 ev to $10^7$ ev at sufficiently close intervals, within the limits of the machine, to resolve as much of the curves as possible; hence the large number of energy points. The curves may be on either a log-log or a linear scale with the log E or E being punched respectively. In the event that a portion of the curve is a straight line, either on a log E or E scale, only the end points are read. If the curve is not given over the entire energy range, that portion which is available is read.

Information cards containing the following information will appear when necessary.

1. The number of cross-section values
2. Abscissa vs log E or E (code number specifying which)
3. Material identification (code number)
4. Material density
5. Type of cross-section (code number)
6. Reference (code number)

The code numbers for the types of cross-sections to be read are listed below:

1. Total - $\sigma_T$
2. Absorption - $\sigma_a$
3. Elastic scattering - $\sigma_{el}$
4. Inelastic scattering - $\sigma_{in}$
5. Total scattering - $\sigma_S$

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(6) Fission - $\sigma_f$

(7) Capture/fission - $\alpha$

(8) Absorption/fission - $1 + \alpha$

(9) Nonelastic - $\sigma_{\text{non}}$

(10) n, gamma reaction - $\sigma_{n,\gamma}$

(11) n, alpha reaction - $\sigma_{n,\alpha}$

(12) n, proton reaction - $\sigma_{n, p}$

(13) n, n' reaction - $\sigma_{n, n'}$

(14) n, 2n' reaction - $\sigma_{n, 2n'}$

(15) n, n' f reaction - $\sigma_{n, n' f}$

The following parameters will be treated as cross-sections.

(16) Flux - $\phi$

(17) Average logarithmic energy decrement - $\xi$

(18) Average cosine of the scattering angle - $\bar{\mu}_L$

(19) Fission spectrum - $F$

(20) Neutrons released per fission - $v$

Since the total and fission cross-sections are the most thoroughly measured, they will always be read. It will not be necessary to read the entire remaining list, since certain items can be computed from others.

B. 1001 ENERGY POINT CROSS-SECTION CODE

The cross-sections read at unequal log E or E spacings must be broken up into the 1000 log E spacings that are listed.

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0.001 \leq E \leq 0.01 \text{ ev} \quad \begin{array}{l} \text{10 equal log E spacings} \\
0.01 \leq E \leq 0.6 \quad \text{90 equal log E spacings} \\
0.6 \leq E < 10^7 \quad \text{900 equal log E spacings} \end{array}

The resulting 1001 cross-sections are to be computed in a manner that will give the same area under the curves as the original curve when plotted on a log E scale. The program then reduces the data to the desired types of cross-sections, including Maxwell-Boltzmann averaged thermal values. It is then able to use this data in an operation to produce fewer-group cross-sections with a flux-weighting option. Both macroscopic cross-sections and material combinations can be handled. Cross-sections such as $\sigma_{el}^*, \sigma_{TR} = (1 - \mu_L) \sigma_{el}$, $\nu \sigma_{tr}$, and D can replace other types, if desired.

A description of the equations, for which coding was begun in FORTRAN, along with the tentative flow diagrams, follows:

1. Phase I

The computations of $\sigma_i$ for each point $U_i = \log E_i$ depend upon whether $i = 0, i = 1000$ or $0 < i < 1000$ (1000 must be replaced by $i_{\text{max}}$ for the case of fewer-group cross-sections). (See Fig. 9)

For $0 < i < 1000$, two possibilities exist. The number of data points within the interval $\frac{U_i - 1 + U_i}{2}$ to $\frac{U_i + U_i + 1}{2}$ plus those coinciding with the end points may exceed zero or it may be equal to zero.

If the number exceeds zero, the two nearest neighbors, one on each side of the upper boundary $\frac{U_i + 1 + U_i}{2}$ must be found. The cross-section at the point, $\frac{U_i + 1 + U_i}{2}$, $\sigma^+$, must then be found by straight-line approximation between these neighbors called $\sigma' (U')$ and $\sigma' (U')$, i.e.,

$$
\sigma_i + \frac{(U_i + U_i + 1)}{2} = \left[ \frac{\sigma' - \sigma''}{U' - U} \right] \left[ \frac{U_i + U_i + 1}{2} - U' \right] + \sigma'
$$
The cross-section at $u_i$ is computed so that 

$$\sigma_i(u_i) \times \text{length of the interval from } \frac{u_{i-1} + u_i}{2} \text{ to } \frac{u_i + u_{i+1}}{2}$$

equals the shaded area in the figure.

**Fig. 9. Various Cross-Section and Lethargy Points**
The cross-section at the lower boundary \( \frac{U_i + U_i - 1}{2} \), \( \sigma^- \), has been calculated:

\[ \sigma^- = \sigma^+ \]

If, in the nearest-neighbor procedure, the U data point coincides with the boundary,

\[ \sigma^+ = \sigma' (U') \]

Sufficient information now is available to perform trapezoidal integration between the above-mentioned boundaries. Dividing by the width of the interval gives: \( \sigma_i (U_i) \).

If the number of data points equals zero, the nearest neighbor on each side of the interval \( \frac{U_i - 1 + U_i}{2} \) to \( \frac{U_i + 1 + U_i}{2} \) must be found. Call them \( \sigma' (U') \) and \( \sigma'' (U'') \). The intersection of the line joining these points with the line \( U = U_i \) is \( \sigma_i (U_i) \):

\[ \sigma_i (U_i) = \left[ \frac{\sigma' - \sigma''}{U' - U''} \right] \left[ U_i - U' \right] + \sigma' \]

For the case where \( i = 0 \), \( \sigma^+ \) must be found by the methods illustrated above. The trapezoidal integration now is between the limits \( U \) to \( \frac{U_0 + U_i}{2} \). Dividing by the width of the interval gives \( \sigma_0 (U_0) \).

For the case where \( i = 1000 \) (or \( i_{\text{max}} \) \( \sigma_i^- = \sigma_i^+ - 1 \)). The trapezoidal integration is now between the limits \( \frac{U_{999} + U_{1000}}{2} \) to \( U_{1000} \). Dividing by the width of the interval gives: \( \sigma_{1000} (U_{1000}) \).

2. Phase II

This phase of the program reduces the many types of cross-sections to the eight desired types. Equations of the following form are employed:
\[ \sigma_a = (1 + \alpha) \sigma_f \]
\[ \sigma_a = \sigma_{n,\alpha} + \sigma_{n, p} + \sigma_{n,\gamma} + \sigma_f \]
\[ \sigma_a = \sigma_T - \sigma_s \]
\[ \sigma_{\text{in}} = \sigma_{n, n'} + \sigma_{n, 2n'} + \sigma_{n, n'f} \]
\[ \sigma_a = \sigma_{\text{non}} - \sigma_{\text{in}} \]
\[ \sigma_{\text{in}} = \sigma_s - \sigma_{\text{el}} \]
\[ \sigma_{\text{el}} = \sigma_T - \sigma_{\text{non}} \]

3. Input

There are three sources of input to Phase I.

(1) Data read from curves--presently those in BNL-325 and UCRL-5226; 2nd edition.

(2) Data resulting from the Breit-Wigner analysis described in the following section.

(3) The output of Phase II.

The first two items when operated upon by Phase I produce input for Phase II. The third item is used when fewer-group cross-sections are desired. The following diagrams show the relationships.

\[ \text{Data read from curves} \rightarrow \text{Phase I} \rightarrow \text{Phase II} \rightarrow \text{1000-group, 8-type* cross-sections} \]

\[ \text{1000-group, 8-type* cross-sections} \rightarrow \text{Phase I} \rightarrow \text{Fewer-group, fewer-type or different-type cross-sections} \]

\[ \ast \sigma_{a}, \sigma_{f}, \sigma_{\text{el}}, \sigma_{\text{in}}, \nu, \mu, \xi \text{ and } F. \]

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All output is stored on magnetic tape with the option of obtaining printed output.

Logical flow charts for Phase I are given in Fig. 10 and flow charts for Phase II are given in Fig. 11.

The FORTRAN programming from these flow charts was started during the quarter. A assembly and checkout will be completed during the next quarter. Details of application will be included in the next quarterly report.

C. USE OF THE BREIT-WIGNER FORMULA

Although a large part of the data necessary to prepare a cross-section library can be found in such publications as BNL-325, UCRL-5226 and UCRL-5351, the interpretation of the resonance region is poor or absent. Since resonance parameters are often given, the Breit-Wigner single-level formula can be used to resolve the experimental total cross-section into the necessary component parts.

The following forms of the Breit-Wigner formula were coded in FORTRAN.

\[
\sigma_{c_i,k} = \left( \frac{A + 1}{A} \right)^2 \frac{2.62 \times 10^6 g \, r^{n_i} \, r^{c_i}}{\sqrt{\frac{E_k}{E_i}} \left[ \frac{(E_k - E_i)^2 + \Gamma_i^2}{r^{n_i} \, r^{c_i}} \right]}
\]

\[
\sigma_{f_i,k} = \left( \frac{A + 1}{A} \right)^2 \frac{2.62 \times 10^6 g \, r^{n_i} \, r^{c_i}}{\sqrt{\frac{E_k}{E_i}} \left[ \frac{(E_k - E_i)^2 + \Gamma_i^2}{r^{n_i} \, r^{c_i}} \right]}
\]

\[
\sigma_{s_i,k} = \left( \frac{A + 1}{A} \right)^2 \frac{2.6 \times 10^6 g \, (r^{n_i})^2}{\left[ 4 \, (E_k - E_i)^2 + \Gamma_i^2 \right]}
\]

\[
+ \left( \frac{A + 1}{A} \right) \frac{6.475 \times 10^3 g \, r^{n_i} \sqrt{\sigma_{PR} (E_k - E_i)}}{\left[ 4 \, (E_k - E_i)^2 + \Gamma_i^2 \right]}
\]

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Fig. 10. Flow Chart for Cross Section Data Program, Phase I
Fig. 11. Flow Chart for Cross Section Data Program, Phase II
Subscripts c, f and s refer to capture, fission and scattering, respectively.

Subscripts i and k refer to resonance and incident neutron energy, respectively.

where:

- \( A \) = atomic mass
- \( g \) = statistical weight factor
- \( E_i \) = resonance energy of \( i \)'th resonance, ev
- \( \Gamma_{n_i}^{o} \) = reduced neutron width of \( i \)'th resonance, \([\text{ev}]^{1/2}\)
- \( \Gamma_{\gamma_i} \) = gamma width of \( i \)'th resonance, ev
- \( \Gamma_{c_i} \) = capture width of \( i \)'th resonance, ev
- \( \Gamma_{f_i} \) = fission width of \( i \)'th resonance, ev
- \( \Gamma_{n_i} \) = neutron width of \( i \)'th resonance, ev
- \( \Gamma_{i} \) = \( \Gamma_{\gamma_i} + \Gamma_{f_i} + \ldots \) (not including \( \Gamma_{n_i} \))
- \( \sigma_{PR} \) = \( 4\pi r^2 \times 10^{-24} \), barns
- \( R \) = \( 1.47 A^{1/3} \times 10^{-13} \) cm
- \( E_k \) = \( k \)'th incident neutron energy, ev

and \( \Gamma_{c_i} \) represents either \( \Gamma_{\gamma_i} \) or \( \Gamma_{p_i} \) where \( p \) denotes particle.

The resulting cross-sections at energy \( E_k \) are given by

\[
\sigma_{c,k} = \sum_i \sigma_{c_{i,k}}
\]
\[ \sigma_{f_k} = \sum_i \sigma_{f_{1,k}} \]

\[ \sigma_{s_k} = \sum_i \sigma_{s_{1,k}} + \sigma_{PR} \]

\[ \sigma_{T_k} = \sigma_{e_k} + \sigma_{f_k} + \sigma_{s_k} \] (inelastic scattering is assumed to be absent in the region of known parameters)

\[ \sigma_{a_k} = \sigma_{c_k} + \sigma_{f_k} \]

In the event that the agreement in the region of known parameters is poor due to unknown resonances at higher energies, an option which adds a correction in this region is available. This correction is given by:

\[ \sigma_{VC_{k}} = \sigma_{oc} \sqrt{\frac{E_0}{E_k}} \]

where \( \sigma_{oc} \) is at energy \( E_0 \), also \( \sigma_{oc} = \sigma_c - \sum \sigma_{c_{1,o}} \), where the sum is over all known resonances and \( \sigma_c \) is the experimental cross-section of \( E_0 \). This correction has a similar form for \( \sigma_{f} \). The scattering cross-section does not obey this \( \frac{1}{v} \) law; hence, the correction does not apply to it.

Since agreement generally is poor near the peak of a resonance, an option which normalizes the calculated total cross-section to the experimental value is available. This normalization factor is applied to each component type of cross-section.

The input to this program requires \( E \) vs \( \sigma_T \) which is available from Phase I of the 1001 Energy-Point Cross-Section Program. The resonance parameters are available in the literature. Output options include either punched cards for input into Phase I of the 1001-Group Cross-Section Program, printed results including the partial cross-sections for each component type or both.

The program was assembled during the quarter and checkout was started. This will be finished during the next quarter and reported in the next quarterly report.

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V. EXPERIMENTAL PHYSICS

In order to provide a reliable basis for the selection of machine programs and to check out the general code, a program of critical experiments is being conducted in the Martin Critical Facility (MCF). The data from this program, supplemented by data from other experimental and operational reactors will establish the accuracy and reliability of APWRC.

In the following sections, a general description of the experimental core and associated equipment is presented. The experimental results obtained during the quarter are then presented along with a preliminary analysis of some of the data. Finally, plans for a future experiment, the homogeneous element substitution experiment, are presented.

A. DESCRIPTION OF THE FACILITY AND EQUIPMENT*

1. Test Cell Equipment

The experimental physics portion of the Code Development Program is conducted in a large 20 x 22 x 20-foot high test cell located in the southeast section of the Martin Critical Facility. The cell is made entirely of poured concrete with two-foot-thick interior and one-foot-thick exterior walls. The reactor core and associated structural and auxiliary equipment are located in an eight-foot-square by twelve-foot-deep pit, as shown in Fig. 12.

The 60-inch diameter stainless steel reactor tank is supported by a five-foot, five-inch-high framework. Inside the reactor tank is a support stand 28 inches high and 45 inches in diameter. The grid structure in which the fuel elements are placed is centered on the support stand. Figure 13 is a drawing of the tank, reactor grid and structural members. Figure 14 is a photograph of the reactor tank with the neutron detectors, mixer and part of the core assembled.

Above the core, reactor tank and test pit is the actuator support structure, a steel framework six feet high, composed of eight-inch I-beams secured to the curb at each corner of the test pit. This upright support holds a series of 8-1/2-foot steel dollies which are used to support and position the various control and safety actuators. The support dollies are placed on two levels, two feet apart, and can be moved easily for centering the actuators over the core. Figure 15 is a front view of the support structure with the various types of actuators mounted. The actuators are placed on small rectangular sleds which can be adjusted to position along the length of the dollies.

*L. Welshans, K. Johnson

MND-C-2200
Fig. 12. Location of Reactor and Structure in the Pit

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Fig. 13. Oblique View Reactor Test Stand Assembly
Fig. 14. View of Reactor Core and Tank
Fig. 15. View of Actuator Support Structure
The pit is coated with an epoxy resin in order to maintain the purity of the demineralized water. A drain in the bottom of the pit is connected to a pump which returns the water to an externally located storage tank. All structural equipment which comes in contact with the water is made of stainless steel, aluminum or coated with plastic.

2. Core Description

The reactor core is made of fuel elements, critical experiment (CE) control rods and safety rods, all of which are positioned in the reactor grid structure. The number of fuel elements will vary according to the core being studied and will range from 9 to 45 elements. Each core will contain two safety rods and usually three CE control rods.

The core is designed so that there are no blocked water channels, thus allowing the moderating water to enter, drain and circulate freely through the core.

Fuel and fuel element assembly.- The fuel consists of strips of fully enriched uranium foil centered in and sheathed between two 0.005-inch-thick stainless steel plates. The stainless steel plates are then seam welded on all four edges to seal in the uranium foil after the plate has been evacuated to a pressure of about 25 microns. The assumed maximum void is 5% of nominal volume.

The dimensions of the uranium foil are 2.5 inches x 22 inches x 0.0018 inch. Each fuel plate contains approximately 28 grams of U-235. The outside dimensions of the fuel plate are 2.725 inches x 23 inches x 0.0118 inch.

The fuel plates are assembled into bundles containing boron stainless steel and stainless steel shim stock. The bundle will contain five layers as follows: stainless steel shim plate; fuel plate; boron stainless steel; fuel plate; stainless steel shim plate. Since the shim plates have the same dimensions as the fuel plate, the bundles are made by placing the various plates on top of each other in the proper order. A strip of mylar tape is attached to each end to hold the plates together.

Each bundle will contain two fuel plates. However, the amounts of stainless steel and boron steel will vary from core to core. This is described later in the experimental program.

MND-C-2200
Then, the bundles are assembled into elements by placing them in stainless steel boxes. Each element contains 18 bundles, which are separated from each other by means of 1/4-inch-wide plastic spacers. The thickness of the spacer will vary as the bundle thickness changes in order to maintain a constant 0.163-inch spacing from the center of one bundle to the center of an adjacent bundle. Figure 16 is a drawing of the fuel element box and illustrates how the bundles are loaded.

The assembled element, containing approximately 1 Kg U-235, is placed in the grid structure where it is held in place by means of lugs on the box which rest on the top of the upper grid. The upper and lower grids space the elements about 2.9 inches between centers of adjacent elements in the horizontal plane and one inch from the grids in the vertical plane. In these experiments, the elements are oriented so that the fuel bundles are in the same or parallel vertical planes.

Safety control rod.- Two safety control rods are used to provide a rapid decrease in the reactor power when the rods are inserted into the core. They consist essentially of four flat plates welded together to form a hollow box which slides between the grid structure. Each plate consists of natural boron powder contained in a stainless steel jacket. The boron content is approximately 30% by volume and the worth of the rod in a large core may exceed 5% reactivity.

During the buildup of a new core, a fuel follower section is attached to the safety control rod. This has the same dimensions, fuel loading and bundle spacing as the previously described fuel element. During normal operations, the fuel in the follower section is in the same horizontal plane as the fuel in the fuel elements. However, when the rod is dropped, the fuel is displaced from the core by the boron section. The resulting large decrease in reactivity produces an additional safety factor during core buildup.

Critical experiment control rods.- These control rods are used for measuring the reactivity changes during the various experiments. They are also used to bring the reactor critical after it is filled with water, since the fuel loading is adjusted so that the reactor is sub-critical critical when the reactor tank is filled with water and the CE control rods inserted in the core. They are made of 0.020-inch-thick stainless steel welded to form a 26 3/4-inch x 2 5/32-inch x 0.10-inch rectangular box. The box is filled with approximately 45 grams of natural $B_4C$ and fitted with a collar for attachment to the actuator drive mechanism. Three CE control rods are usually required in a core in order to completely calibrate the rods.
Fig. 16. Standard Fuel Element
The control rod actuators are rack-and-pinion reversible drive actuators with a maximum speed of 12 inches per minute, a 22-inch travel length and a reproducible positioning accuracy of ±0.01 inch. They limit the rate of increase of reactivity by control rods to a maximum of 0.006% reactivity per second.

Limit switches are mounted on the actuators to stop the drives automatically when the rod is fully withdrawn from or fully inserted into the core.

3. Water System and Temperature Control

Demineralized water is stored in a 3000-gallon tank located outside the test cell. The tank is lined with sheets of sulfonated polyethylene to prevent contamination of the water. The storage tank is provided with electric heaters to prevent freezing during the winter. The water flows by gravity to the reactor tank through a four-inch stainless steel line containing two air-operated valves. One valve is the normal fill valve while the other is a back-up in case the fill valve malfunctions.

The fill valve closes automatically when the water level in the reactor tank reaches 66 inches. This provides approximately six inches of water as a reflector region above the core.

A six-inch valve at the bottom of the reactor tank provides the means of dumping the water into the pit in one minute.

The water is heated in the reactor tank by means of steam immersion heaters at the rate of 1°F per minute. Circulation is provided by three mixers.

Reactor water temperature is measured by three platinum resistance bulbs located outside the core. In addition, five iron-constantan thermocouples measure the water temperature at various points within the core. Relative temperature differences of the water can be measured within 0.1°C.

4. Reactor Instrumentation

A block diagram of the reactor instrumentation is shown in Fig. 17. Channels 1 and 2 are startup channels, used mostly to determine multiplication factors during buildup of new cores. Channels 3 and 4 are the normal operating instruments for determining criticality. Channel 5 provides period and power indication, while channel 6 is the gamma level outside the core. Channel 7 is an alternate operating channel that does not contain any reactor scram function, as do channels 3, 4, 5, and 6. A photograph of the complete console is shown in Fig. 18.
Fig. 17. Block Diagram of Reactor Instrumentation

MND-C-2200
Fig. 18. Reactor Console
A single channel analyzer, channel 8, can be used for both neutron and gamma spectrum analyses.

Other non-reactor instrumentation which is mounted in the console includes a television receiver and an intercommunication system. An aural monitor provides a signal which can be heard in the test cell and the control room. The signal rises in frequency with increasing reactor neutron level.

B. EXPERIMENTAL PROGRAM

Critical experiments were performed on two core configurations during this reporting period. The first assembly, designated Core 403, had been investigated in a previous series of experiments. (See Ref. 7.) The assembly was rearranged to remove certain perturbations from the core and a short series of experiments was performed. A preliminary analysis of the experimental data was also performed. The second core assembled during the report period was designated Core 454. Experimentation on this core was nearing completion at the end of the quarter.

1. Critical Experiments on Core 403*

Fuel bundle assembly.- The fuel bundles for the 403 core were assembled of plates as follows: 5-mil stainless steel shim stock; 11.8-mil fuel plate; 4.2-mil boron steel; 11.8-mil fuel plate; 5-mil stainless steel shim stock. The final bundle thickness was 37.8 mils. The plastic spacers were 122 mils in thickness; hence, the total cell size was 159.8 mils thick. Figure 19 is a drawing of a typical fuel bundle with nominal dimensions shown. While it is theoretically possible to construct a 163-mil-thick cell in the standard fuel element, dimensional tolerances and compressibility of the bundles limit the cell to thicknesses of 159 to 161 mils.

Critical mass.- The previously constructed critical core was rearranged to a 4 x 4 array with the safety rods placed on the outside of the core without their fuel follower boxes. The usual CE control blades were replaced by a control rod which was inserted in the water channel between bundles in an element. Thus, when the safety rods were cocked and the CE control blade withdrawn, the core consisted solely of standard fuel elements, each containing 18 bundles. The critical configuration of this core is shown in Fig. 20, along with the location of the various sensors. The cold, clean critical mass at 22.3° C was found to be 15,560 grams of U-235, as compared to the previous critical mass of 15,700 grams.

*L. Welshaus, K. Johnson

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NOTE: 1. All dimensions in inches

Fig. 19. Typical Fuel Element Bundle for 403 Core
Fig. 20. Cold, Clean, Critical Configuration--Core 403
Buckling measurements.- Three techniques were used to measure transverse buckling in the above core. The transverse direction is defined as being perpendicular to the plane of the fuel plates. Critical bucklings were measured by:

1. Fuel plate scanning
2. Gold foil activation
3. Thorium foil activation.

The fuel plate scanning technique consists of placing unexposed fuel plates in the reactor in the positions shown in Fig. 21. The reactor was brought critical at a power of two watts for a period of 20 minutes. The gamma radiation emitting from the center of the fuel plate was then detected by a scintillation counter placed above a 1/4-inch hole in a 4-inch-thick slab of lead. The data was fitted to a cosine curve by means of a machine program which is described in the following section. The machine fit data is shown in Fig. 22 where the solid curve is the best fit equation and the circles represent experimental data. Since there was a CE control rod, inserted to 10.22 inches, adjacent to element 45, the data on the right side of the curve showed more scatter. As a result, the equation of the best fit curve was machine-derived, using data from the edge of the core to the center. The resulting curve, Fig. 23, shows somewhat less scatter of the data and a decrease of approximately 0.5 cm in the reflector savings.

The technique appears to give satisfactory results and the accuracy of the measurements will be improved by modifications of the fuel plate positioner and scintillation counting equipment which are currently underway.

Two-mil-thick by 11/16-inch diameter gold foils were irradiated in the positions indicated in Fig. 21. The foils were taped to a fuel plate with Mylar tape. The experimental data along with the machine fit cosine curve are shown in Figs. 24 and 25. The experimental data presented in Fig. 24 was taken at the center of the fuel plate (11-1/2 inches from the bottom). The data shows considerable scatter; however, the chief objection is the lack of sufficient data points. At least 12 to 16 points should be taken over the center half of the core to provide sufficient data for a good cosine fit.

The data shown in Fig. 25 was taken at 9 inches from the bottom of the fuel plate in order to decrease the effect of the partially inserted control rod.
CE control rod inserted 10.22 in.

Right hand number represents bundle number used in fuel plate scanning.

Center number represents foil positions.

Not to scale.

Fig. 21. Core 405 Showing Fuel Plate and Foil Locations for Buckling Measurements.
Flux = \cos 0.07856 (x - 14.90)

Fig. 22. Core 403 Buckling Measurement (Fuel Plate Scanning, Transverse Direction)
Fig. 23. Core 403 Buckling Measurements--Fuel Plate Scanning (Transverse Direction--One-Half Core Distance)

Flux = cos 0.073745 (x - 15.82)
Flux = \cos 0.07557 (x - 15.30)

Fig. 24. Core 403 Buckling Measurement, Gold Foil--Run No. 1 (transverse direction 11.5 in. from core bottom)
Flux = \cos 0.0732 (x - 14.51)

Fig. 25. Core 403 Buckling Measurement, Gold Foil--Run No. 2 (transverse direction 9 in. from core bottom)
The thorium measurements were made with five-mil-thick by 15/16-inch diameter thorium metal foils. These were taped to the fuel plates in the same positions as the gold foils. Differential pulse height analysis of the foil activity was determined using the physics measurement channel as a single channel analyzer (Channel 8 of Fig. 17). Two gamma peaks were found at 0.098 and 0.66 mev, due to the n, γ reaction. The discriminator was set at 1.1 mev to eliminate these gammas and record only fission-product gammas. Since the fission threshold is above 1 mev, the foils can be used as fast neutron detectors.

A decay curve was prepared at the above discriminator setting in order to correct for foil decay. Figure 26 is a plot of the best decay curve. This indicates that the mean half-life of the fission products being counted is approximately 17 minutes. Hence, the measurements must be completed within an hour to obtain reasonable statistics.

Figure 27 is a plot of the first determination along with the machine fit curve. The data on the right side of the curve shows considerable scatter, due to the presence of the adjacent CE control rod. In this experiment, the foils were located 11-1/2 inches from the bottom of the fuel plate. A better fit was obtained (Fig. 28) when the foils were located 9 inches from the bottom of the fuel plate, at which distance the flux was less perturbed by the partially inserted control blade.

The thorium foils show less peaking in the reflector region than the gold foils. This indicates that the thorium foils can be used as fast neutron detectors.

In addition to the above three methods, buckling measurements were made with indium and dysprosium foils. The data is not presented here, since it is necessary to calibrate the foils individually before the data can be fit to a cosine curve. However, these thermal flux detectors appear to be strongly influenced by the change in thermal flux at the boundaries between adjacent fuel elements.

The chief result from the experiments on Core 403 was the development of methods for measuring critical buckling. This core will be reassembled later with the fuel and boron distributed uniformly from element to element. At that time the buckling measurements will be made along the three axes of the core.

2. Comparison of Experimental and Analytical Results for Core 403.*

In general, comparison of experimental and analytical results will be postponed until completion of the pilot code described in Chapter III. The use of automated analytical techniques will greatly reduce both

*L. Welshaus, R. Hubner
Fig. 26. Thorium Activity Decay Curve

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Flux = \cos 0.07603 (x - 13.46)

Fig. 27. Core 403 Buckling Measurement, Thorium Foil, Run No. 1,
Transverse Direction--11.5 in. from Core Bottom
Flux = \cos 0.07257 (x - 14.286)

Fig. 28. Core 403 Buckling Measurement, Thorium Foil, Run No. 2, Transverse Direction--9 in. from Core Bottom
machine time and total span time for these comparisons. However, it was decided to analyze the buckling measurements in order to select the most desirable foil materials and to refine the experimental techniques which will be used on future cores.

Results of measurements of the flux-versus-position and power-versus-position through Core 403 perpendicular to the fuel bundles using thorium and gold foil activation and fuel plate scanning respectively were analyzed, using COFIT, (Ref. 8), a least-squares cosine fitting program, to obtain the experimental buckling and reflector savings. This program fits the equation \( y = A \cos (BX - C) \) to the data. The data and cosine curves are reported in the previous section of this report.

A comparison with the analytical buckling and reflector savings as computed by C3, F3 and ELM theory, is given in Table 4.

**TABLE 4**

Results of Analysis of Experimental Activation Data

<table>
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<tr>
<th>Experimental</th>
<th>Radial Reflectors</th>
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</thead>
<tbody>
<tr>
<td>Method</td>
<td>Savings * (cm)</td>
</tr>
<tr>
<td>Thorium activation (fast)</td>
<td>3.40 ± 0.28</td>
</tr>
<tr>
<td>Fuel plate scanning (fission)</td>
<td>4.47 ± 0.27</td>
</tr>
<tr>
<td>Gold activation (thermal)</td>
<td>6.44 ± 0.37</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Analytical</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fast B^2, cm^-2 Epithermal B^2, cm^-2 Thermal B^2, cm^-2</td>
</tr>
<tr>
<td>0.00548 0.005245 0.004607 0.005014</td>
</tr>
<tr>
<td>3.576 1.060 0.3062 0.087</td>
</tr>
<tr>
<td>0.038 0.586 0.376 1.000</td>
</tr>
</tbody>
</table>

\[ \lambda_{TR} = 0.71 \]  

Radial Reflectors savings, cm  

*Core dimension = 29.41 cm, extrapolation distance = 3.58 cm; thorium = 0.87 cm; fuel plate scanning = 0.31 cm; gold

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The dysprosium data was similarly analyzed, using approximate foil calibrations. Agreement between analysis and experiment, in this case, was poor. Evidently, this is due to the small number of measurement points, an apparent flux perturbation due to partial insertion of CE rods and inexact foil calibration. For the data in Table 6, good agreement is seen for the reflector savings only for thorium and the $B^2$ agreement is, in general, only fair. An improvement in the experimental techniques which will occur for future measurements improve the agreement for fuel plate scanning, dysprosium and gold.

Another probable cause of disagreement is in the nature of the comparison. The analytical reflector savings was obtained by the ELM theory by calculating the neutron buckling and leakage in each energy group and then finding the size of a bare core, of the same material, which has the same total leakage. To perform this calculation, using the experimental bucklings in Table 4, one must have the absolute, or at least the relative, flux in each energy group. An attempt will be made to measure these during the next set of experimental buckling measurements.

Until this is done and until the flux measurement techniques are improved, it is evident that the thorium technique gives the best agreement between analysis and experiment for the present.

3. Critical Experiments on Core 454 *

Fuel bundle assembly.- The fuel bundles for Core 454 were assembled to a thickness of 43.4 mils. The composition of each bundle is as follows: 7-mil stainless steel shim stock; 11.8-mil fuel plate; 5.8-mil boron steel; 11.8-mil fuel plate; 7-mil stainless steel shim stock. Figure 29 is a drawing of a typical fuel bundle with nominal dimensions shown. The boron steel was selected so that each element contained 770 ± 2 grams of boron steel.

Composition of a typical element.- Each element incorporates 18 fuel bundle assemblies and contains approximately one kilogram of U-235 (93.17% enrichment). The average composition of an element is as follows:

(1) Stainless steel 6067.69 grams
(2) Natural boron (based on analysis of 1.36% boron) 10.40 grams

*L. Welshaus, K. Johnson
NOTE: All dimensions in inches

Fig. 29. Typical Fuel Element Bundle for 454 Core
The average composition of a fuel bundle is as follows:

1. Stainless steel 322.72 grams
2. Natural boron 0.578 grams
3. U-235 (93.17% enrichment) 55.3 grams
4. Lucite 693.86 grams

Critical core buildup. - The location of the neutron detectors and source prior to the initial fuel addition is illustrated by Fig. 30. Before the multiplication loadings were initiated, a background count and source term were obtained. The background count was zero. The source term was determined with the reactor tank full of water and the ten-curie Po-Be source in the core. Several one-minute counts were obtained. From these counts an average source term was calculated. Criticality was approached by the incremental addition of fuel as shown by the loading sequence diagram (Fig. 31). The number in each box indicates the run on which that particular element was added to the core. At each loading, several counts were obtained (on both startup channels) with the CE control rods fully inserted and fully withdrawn from the core. The average inverse multiplication (ratio of source term to count rate) for each channel was plotted as a function of kilograms of U-235. Figure 32 shows the inverse multiplication curve with the multiplications at each point averaged for both channels. The core achieved criticality at a temperature of 24.5°C when it contained 24.9 kilograms of U-235 in the configuration shown in Fig. 33. CE rod one was completely out of the core and rod two was inserted to 15.14 inches. To determine the exact critical mass, four partial fuel removal runs were completed. The critical mass of the 43.4-mil core was found to be 24,507.75 grams of U-235 and 260.67 grams of natural boron (based on 1.36% boron in the steel) at a temperature of 23.6°C with the CE control rods fully withdrawn from the core. Figure 34 illustrates the critical configuration. The safety rods were positioned outside the active core with their fuel follower sections removed.

Experimental core buildup. - After determining the cold, clean critical mass, the core was built up to the experimental configuration. The buildup was accomplished by the addition of another CE control rod along with additional fuel. Adjustments between fuel and CE control rods were made until the desired configuration, as shown by Fig. 35, was obtained. The experimental core achieved criticality at a temperature of 23.5°C, containing 24.62 kilograms of U-235; the CE control rod positions were:

MND-C-2200
Fig. 30. Location of Neutron Detectors and Source
Fig. 31. Sequence of Initial Criticality Loading--Core 454
Fig. 32. Inverse Multiplication Curve for Initial Loading of Core 454
<table>
<thead>
<tr>
<th></th>
<th>62</th>
<th>52</th>
<th>42</th>
<th>32</th>
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<td>43</td>
<td>33</td>
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<td>64</td>
<td>54</td>
<td>44</td>
<td>34</td>
<td>24</td>
<td>CE-2</td>
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<td>67</td>
<td>57</td>
<td>47</td>
<td>37</td>
<td>27</td>
</tr>
</tbody>
</table>

Fuel Plate Orientation

Fuel in First Bundle of Elements 27-37-47-57-67

**Fig. 33. Initial Critical Core Configuration--Core 454**
Fig. 34. Cold, Clean Critical Core Configuration--Core 454

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Fig. 35. Experimental Core Configuration--Core 454

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Rod No. 1--14.25 in. withdrawn (7.75-in. inserted)

Rod No. 2--100% inserted in core

Rod No. 3--100% withdrawn from core.

Calibration of the CE control rod system.- The three CE control rods in the experimental core (Fig. 35) were calibrated by means of incremental period measurements and subcritical multiplication techniques. The agreement in rod shapes by the two methods is good. The calibration curve for rod No. 1 showing percent of total rod worth versus rod position is presented in Fig. 36. The curves for the other two rods are similar.

The total worth of the CE control rod system is 0.315% $\frac{\Delta k}{k}$. This is distributed among the three rods as shown below:

<table>
<thead>
<tr>
<th>CE Rod Numbers</th>
<th>Reactivity Worth % $\frac{\Delta k}{k}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.082</td>
</tr>
<tr>
<td>2</td>
<td>0.144</td>
</tr>
<tr>
<td>3</td>
<td>0.089</td>
</tr>
</tbody>
</table>

The values of rods 1 and 3 are similar, since they are located in symmetrical positions in the core.

Fuel and boron worth evaluation.- This evaluation was performed for the purpose of determining the fuel and boron worths in various bundle locations in the core. The fuel bundle nearest the center of the element (position No. 9 of each box) was chosen as the experimental location for each evaluation. The initial evaluation was made in the quadrant of the core illustrated by Fig. 37. For each measurement obtained in each specified bundle location, the appropriate fuel and boron was replaced with 12- and 6-mil stainless steel strips, respectively.

Figures 38 and 39 indicate the absolute reactivity worth of each fuel and boron measurement performed in the experimental quadrant of the core.

The fuel and boron worths are summarized in Tables 5 and 6. Worths are expressed in percent reactivity per gram of U-235 or per gram of natural boron. Tables 5 and 6 present the values measured and also the measurement number which identifies the data points in Figs. 40 and 41.
Fig. 36. Percent Rod Worth as a Function of Position--CE Control Rod No. 1--Core 454

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Fig. 37. Experimental Quadrant for the Fuel and Boron Evaluation--Core 454

MND-C-2200
Values Given in $\frac{\Delta k}{k}$ per Gram of $^{235}U$

Top View of Core

Fig. 38. Actual Fuel Worth--Core 454

MND-C-2200
<table>
<thead>
<tr>
<th></th>
<th>62</th>
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<td>66</td>
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</tbody>
</table>

Values Given in $\% \frac{\Delta k}{k}$ per Gram of Natural Boron

Top View of Core

Fig. 39. Average Boron Worth--Core 454
Fig. 40. Radial Fuel Worth--Core 454
Fig. 41. Radial Boron Worth--Core 454
An examination of the results in Tables 5 and 6 shows that the fuel and boron worths are of the expected magnitude. That is, the fuel and boron reactivity worth diminished as the radial distance (from the center of the core) for each evaluation increased. The average radial fuel and boron worth presented in Figs. 40 and 41 respectively, is simply the best curve fit to all of the various data points.

TABLE 5
Fuel Worth

<table>
<thead>
<tr>
<th>Measurement Number</th>
<th>Box Number</th>
<th>Radial Distance (inches)</th>
<th>Percent Reactivity per Gram U-235 $\left(% \frac{\Delta k}{k} \text{ per gram}\right)$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>44</td>
<td>0</td>
<td>+0.003262</td>
</tr>
<tr>
<td>2</td>
<td>45</td>
<td>3.0</td>
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<tr>
<td>3</td>
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<td>9</td>
<td>66</td>
<td>8.46</td>
<td>+0.000446</td>
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TABLE 6
Boron Worth

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<thead>
<tr>
<th>Measurement Number</th>
<th>Box Number</th>
<th>Radial Distance (inches)</th>
<th>Percent Reactivity per Gram Boron $\left(% \frac{\Delta k}{k} \text{ per gram}\right)$</th>
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<tr>
<td>10</td>
<td>44</td>
<td>0</td>
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<td>11</td>
<td>45</td>
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<td>14</td>
<td>55</td>
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<td>-0.05702</td>
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MND-C-2200
TABLE 6 (continued)

<table>
<thead>
<tr>
<th>Measurement Number</th>
<th>Box Number</th>
<th>Radial Distance (inches)</th>
<th>Percent Reactivity per Gram U-235 ($% \frac{\Delta k}{k}$ per gram)</th>
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</thead>
<tbody>
<tr>
<td>15</td>
<td>56</td>
<td>6.7</td>
<td>-0.03352</td>
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<td>16</td>
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<td>18</td>
<td>66</td>
<td>8.5</td>
<td>-0.01044</td>
</tr>
</tbody>
</table>

Examination of the radial fuel and boron worth curves shows that the coplanar worths are higher than the corresponding transverse worths. This would be expected, since the coplanar flux is higher in magnitude than the transverse flux, as shown from buckling.

Up to this point, all of the measurements had been made in one quadrant of the core. To check the relative fuel and boron values between two symmetrical elements, such as 52 to 56 and 26 to 66, experimental measurements were conducted in the same manner as previously described. That is, each evaluation was performed in position 9 in the fuel element. The fuel and boron worths are summarized in Table 7.

TABLE 7
Fuel and Boron Reactivity Worths in Symmetrical Positions

<table>
<thead>
<tr>
<th>Bundle Location</th>
<th>Weight Boron (grams)</th>
<th>Reactivity Worth per Gram ($% \frac{\Delta k}{k}$ per gram)</th>
<th>Weight U-235 (grams)</th>
<th>Reactivity Worth per Gram ($% \frac{\Delta k}{k}$ per gram)</th>
</tr>
</thead>
<tbody>
<tr>
<td>52</td>
<td>0.5821</td>
<td>-0.02577</td>
<td>52.02</td>
<td>0.000923</td>
</tr>
<tr>
<td>56</td>
<td>0.5668</td>
<td>-0.03352</td>
<td>56.49</td>
<td>0.000832</td>
</tr>
<tr>
<td>26</td>
<td>0.5800</td>
<td>-0.00862</td>
<td>52.50</td>
<td>0.000514</td>
</tr>
<tr>
<td>66</td>
<td>0.5746</td>
<td>-0.01044</td>
<td>51.60</td>
<td>0.000446</td>
</tr>
</tbody>
</table>

MND-C-2200
These measurements show that the symmetrical evaluations are in fair experimental agreement. It is felt that better results could be obtained by distributing the fuel more uniformly throughout the core.

Critical buckling measurements. - Critical buckling measurements were made using gold and thorium foils positioned along the three axes of the core. Foil locations for the three traverses are indicated in Fig. 42.

The gold foil measurements are presented in Figs. 43, 44 and 45. It should be noted that the solid curves in these illustrations and the expressions for the fluxes are the results of hand fits to the experimental data. The data was obtained near the end of the quarter and machine fits have not yet been obtained.

The flux as determined with thorium foils was measured using a different technique from that described previously. Two foils, designated monitor foils, were used to control the counting rate of the scalers and thus provide an automatic correction for radioactive decay of the foils. The procedure is to calibrate two scintillation counters and set the discriminators at 1.02 mev base line. The monitor foils were counted with one detector while the remaining foils were counted in the second detector. When the count of the monitor foils reached a preset value of 10,000 counts, the counting was stopped on the other scaler.

The data from these measurements will be presented in the next quarterly report.

C. PRE-EXPERIMENT PLANNING

The overall experimental effort for the Code Development Program was outlined in Ref. 6. Additions or revisions to this effort will be made as the program progresses and the need for specific experimental data becomes evident. During the previous quarter, plans for the homogeneous element substitution experiment were formulated. It was decided that two experimental techniques would be utilized.

The first technique, called simultaneous homogenization, is the substitution into a heterogeneous reference core of a fuel element with the same material concentration as in the reference element but with the materials homogenized in a plastic matrix with the same outside dimensions as a fuel element. The observed reactivity change is used to calculate the critical mass of the core with all homogeneous elements. The difference in critical mass is attributed to heterogeneity of the reference element. The technique involves analytical extrapolation of the experimental reactivity change, according to the theory described in Ref. 9 and is therefore semiempirical.
Fig. 42. Foil Positions for Buckling Measurements—Core 454
Flux = cos 0.0615 (x - 18.54)

Fig. 43. Core 454 Transverse Buckling Measurement, Gold Foil—Run No. 1
(Hand Fitted Curve)
Fig. 44. Core 454 Coplaner Backling Measurement, Gold Foil--Run No. 1 (hand fitted curve)
Flux = \cos 0.0452 (x - 29.21)

Fig. 45. Core 454 Axial Buckling Measurement, Gold Foil, Run No. 1 (hand fitted curve)
The second technique is called individual homogenization. In this case, the boron or uranium content of a reference element is homogenized separately, e.g., when the boron is homogenized, the stainless steel and uranium are retained in heterogeneous, flat plate form. The homogenized material is dispersed in a plastic matrix and inserted into the water gap between bundles. Hence, the cell correction for the homogenized material is one. The experimental reactivity resulting from substituting an off-reference element is observed. Reference 9 theory can again be applied to find the additional amount of each material required to maintain criticality in the presence of heterogeneity. Of course, the technique again requires extrapolation and is semiempirical.

The errors involved in the extrapolation, using either technique, are probably second-order and do not render the results invalid. In addition, the experimental substitution of one element can be duplicated analytically without performing an extrapolation. In this case, it would be assumed that the cell corrections were correct if the analytical and experimental reactivity changes agreed.

Without any additional experimental effort, the extrapolation can be performed analytically with the realization of the first-order limitations on the accuracy. The resulting semiempirical critical mass of each material can be used to mockup the partially or completely homogenized critical core analytically. If the analytical reactivity is zero, the cell corrections may be assumed correct to first order.

In this case of analytical mockup of extrapolated critical mass, the simultaneous homogenization technique has the advantage of requiring no cell correction calculations, since they are all one. For the individual homogenization technique they are one for only a single material at a time.

Simultaneous homogenization experiments will be performed on the following six cores:

(1) Core 404
(2) Core 403
(3) Core 402
(4) Core 354
(5) Core 353
(6) Core 352.

Individual homogenization experiments will be performed on Cores 404, 403 and 402.

Specifications for the homogeneous elements were in preparation at the end of the quarter and will be presented in the next report.
REFERENCES


