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Nuclear parameter study in one, two, and three dimensions

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NUCLEAR PARAMETER STUDY IN ONE, TWO, AND THREE DIMENSIONS

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ABSTRACT

A parameter study comparing one-, two-, and three-dimensional reactor design analyses was undertaken. Computer programs (codes) were used extensively for the computations and running times on the IBM 360/50 were observed. The values of primary concern were neutron flux shapes, effective multiplication factors, and fuel depletion.

Dresden I nuclear power plant with fuel type 1 represents the model reactor. Cross section calculations were performed using the nuclear codes HRG and TEMPEST II, FEVER, EXTERMINATOR-2, and FLARE generated the nuclear parameters.

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TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

I. INTRODUCTION

The objective of this investigation is the analysis and comparison of reactor design parameters calculated by one, two, and three-dimensional methods. The basic procedure entails selection of a model power reactor and a set of nuclear codes that are capable of calculating neutron cross sections and design parameters in one, two, and three dimensions. Dresden I was chosen as the model reactor since it satisfied the required criteria and the reactor core description was obtainable. Selection of the codes was based on their ability to execute the desired calculations. However, in some cases, compromises had to be made due to computer incompatibility and unavailability of codes. The nuclear codes* adopted include the cross section codes HRG and TEMPEST-II and the diffusion codes FEVER, EXTERMINATOR-2, and FLARE.

Correlation of elapsed computer time and nuclear parameter accuracy is an important area in fuel analysis and design. Industrial concerns are particularly interested in using the fastest and most economical design methods available and at the same time maintaining a suitable level of reliability. Hopefully, this would help alleviate overdesign in power plant and fuel analysis resulting in considerable financial savings.

In addition to its direct application to the nuclear industry, the selection of this investigation was complimented by the author's experience with computer operations. Much valuable experience was also gained by working with and understanding the workings of several nuclear codes.

* Discussed in the Literature Survey

II. LITERATURE SURVEY

A. APPLICATIONS AND DISTINCTIONS OF NUCLEAR CODES

One of the most important tools available for reactor design studies today is the electronic digital computer. With it one is capable of performing sophisticated calculations which aid in optimum core design, isotope change and maximum fuel life determinations, and improved safety techniques. The emphasis in this investigation will be placed on criticality and fuel burnup studies.

A large commercial power reactor must be economical as well as safe. Economy is strongly influenced by the precision with which nuclear and thermal hydraulic parameters can be predicted. The type and accuracy of the fuel burnup analysis are also important. In fuel burnup alone there may exist a 10-20%[1] uncertainty in the initial prediction of end of life k_{eff}, which results in an added half to a million dollars in the annual fuel cost of operating a 500-MWe plant. However, as the core life increases, the predictions may be updated by the operational history of the reactor, lessening the full impact of the initial uncertainty. A feedback system between theoretical approximations and actual results would continually improve the computational methods available.

Of primary concern to the reactor purchaser as well as the designer and manufacturer is how temperature, pressure, and time affect the criticality and core power distribution over lifetime. Most studies which incorporate the above analysis can be categorized according to the number of space dimensions which they account for

explicity. The space dimensions, e.g., cartesian-XYZ, cylindrical-RθZ, or spherical RθØ, are important when solving the neutron diffusion or transport equations in reactor analysis. If a code uses only one dimension (X) in its solutions, it is termed a one-dimensional code in contrast to a two-dimensional (XY) or three-dimensional (XYZ) code. Most of these codes have been developed through industrial, government, or university research since their size and complexity can become quite involved. The size of each code is dependent almost entirely on three quantities: (1) number of energy groups, (2) number of regions or compositions, and (3) number of mesh or space points.

B. ONE-DIMENSIONAL METHODS

Of the one-dimensional codes, some solve the neutron diffusion equations[2,3,5,6,7] while others use a more detailed approximation to the neutron transport equation.[4,8,9] The number of energy groups which they are capable of handling ranges from one to as many as thirty[6] and more. The maximum number of regions and mesh points allowed varies from twenty[3] regions and ten[2] space points to fifty[7] regions and 150[6] mesh points. Some codes place a limit on a combination of energy groups and mesh points thus making their limits interdependent. Other programs contain a variable dimensioning technique which allows the program to use only the storage required for each particular problem. This also permits easy adoption of codes for use on other computers.

The more groups, regions and space points a code uses, the more accurate the results. However, the computer time used and hence cost increases.

Other distinctions include the number of energy groups to which downscatter or upscatter is allowed. FEVER[3], for example, allows downscatter to only the next lower energy group whereas SIZZLE[5] allows scatter to a maximum of five successive lower groups.

The cross sections required by the codes generally consist of the microscopic absorption, transport, fission, and scattering cross sections for each nuclide and group. The macroscopic cross sections in most cases are computed internally using the microscopic cross sections and the number densities that are input. Slabs, cylinders, and spheres can be handled by most one-dimensional codes.

Fuel burnup has been included in only a few one-dimensional codes. SIZZLE[5] and DTF-BURN[8] are capable of such calculations which determine the number densities, multiplication factors, and other parameters as a function of time. FEVER[3] and LASER[9] perform a control-poison search and adjust the number density of the control poison (simulated control rod movement) to keep the reactor critical as burnup progresses.

C. TWO-DIMENSIONAL METHODS

A large number of two-dimensional codes are in existence since

they are useful in providing more detailed information and give a more realistic picture of the power distribution in a reactor core. Even though two space dimensions, e.g., XY, RZ, or Rθ, are incorporated, the elapsed time per problem is generally not unreasonable. The complexity of calculations and available options encompasses a broad area.

Energy groups are more restrictive ranging from a maximum allowable of two[10] to a maximum of fifty[13,16]. One collection of programs[10,11,13,16] uses the Equipoise method[19] which is a simple iterative procedure for group-diffusion calculations. No proof for convergence is available but the authors' experience with the method has been quite good. Again, there are those two-dimensional codes which solve the neutron diffusion equations[10,11,13,14,15,16,17,18] and those that solve the neutron transport equation[12].

In some cases, the running times for two-dimensional codes is rather unpredictable due to difficulty in reaching convergence of the eigenvalue (k_{eff}) and neutron flux. Convergence is consistently defined as, e.g., flux convergence

$$[\phi^{n+1}-\phi^{n}]_{max} / \phi^{n} \leq \varepsilon$$

where Ø is the neutron flux at some point, n is the index of the iteration, and ε is some small previously defined value (10^{-4}).

Two-dimensional burnup codes[14,16,18] are also common and often an extension of a previously written two-dimensional diffusion code.

D. THREE-DIMENSIONAL METHODS

Three-dimensional codes are not exceedingly popular mainly because of the tremendous number of mesh points that must be handled and the long computer times required. The three-dimensional mesh point arrays occupy much computer memory leaving less room for the number of allowable nuclides, energy groups, etc. WHIRLAWAY[21], which is a three-dimensional code using the Equipoise[19] method for solution of the diffusion equations, can handle only two energy groups. Working in XYZ geometry, IBM 7090 running time is approximately two to three hours for a 20x20x25 mesh. Importance of detail must be considered to justify this quantity of computer time.

The FLARE[22] code uses a somewhat different approach and is described in Sec. III-B-4. It is used extensively in the ISOCHECK[20] method for determining analytically the amount of isotopes at any time in a particular reactor. Several power reactors have been analyzed using the ISOCHECK method. Among them was DRESDEN I.

A code which is capable of working diffusion problems in one, two, and three dimensions is PDQ7[23]. It can handle rectangular, cylindrical, or spherical geometry in one dimension, the same in two dimensions with hexagonal geometry replacing spherical, and in three dimensions rectangular and hexagonal geometries are available. Three point, five point, and seven point difference equations are used in solving the one, two, and three dimensional cases. Fuel depletion is also provided as an option.

E. CROSS SECTION CALCULATIONS

As part of the input for the above mentioned static design codes, a detailed knowledge is needed of the neutron cross sections for the nuclides that appear in the analysis. The cross sections involved include absorption, transport, fission, and up and/or downscatter depending on the code. These detailed cross sections are condensed over an energy range to produce region flux averaged values, such as

$$\bar{\sigma}_x = \int_{E_1}^{E_2} \phi(E)\sigma_x(E)\,dE/\Phi$$ where E_1 and E_2 represent limits of condensed energy group and Φ equals the total flux over E_1-E_2 range.

Nuclear constants for the diffusion equations are computed using these averaged or broad group cross sections. Computer codes which are capable of performing this condensing technique have been developed for a given range of energies and in some cases for a given region of interest in a reactor. Two logical classifications for these codes are fast and thermal cross section computations.

1. Fast Cross Sections

The principle codes used for high energy (>0.7ev) neutron cross sections are GAM1[24], HRG[26], and FORM[25]. FORM condenses 54-group microscopic cross sections into a few broad groups the number and energy limits being specified by the user. It works with homogeneous media but contains a heterogeneous correction. The GAM1 code is similar to FORM assuming an infinite lattice and a homogeneous medium. It contains a 68-group library and accounts for resonance absorption in Th^{232} and U^{238}.

HRG is identical to GAM1 except for two improvements: (1) it contains a more extensive library, and (2) resonance absorption in more nuclides is considered.

2. Thermal Cross Sections

THERMOS[28] and TEMPEST II[27] are the principle thermal energy cross section codes. THERMOS works with an infinite lattice of heterogeneous fuel cells. It solves the integral transport equation with isotropic scattering to obtain the neutron flux spectrum and thus computes space-energy averaged cross sections. On the other hand, TEMPEST assumes an infinite homogeneous medium and bases its neutron flux spectrum upon the Wilkins equation (heavy moderator), Wigner-Wilkins equation (light moderator), or the Maxwellian distribution.

Both fast and thermal cross section calculations have been incorporated into some programs. One of these codes LASER[9] combines the slowing-down program MUFT and the thermalization transport theory program THERMOS to obtain the neutron spectrum and condensed energy group cross sections. However, its output includes only three-group cross sections and excludes transport or scattering. GAMTEC II[29] encompasses the energy range from 0 to 10 MEV by combining the TEMPEST II and GAM1 codes with an improvement on the resonance absorption contribution to the multigroup constants. It can accommodate either homogeneous or heterogeneous fuel cells. Difficulties have been encountered with its operation at both the University of Missouri - Rolla and Purdue University.

III. DISCUSSION

A. DESCRIPTION OF DRESDEN I REACTOR

The first step in the investigation is to describe the Dresden I core and extract the needed information for the comparative study. Normally in designing a reactor there exist only tentative draft-board dimensions and values which may be adjusted to meet the desired design specifications. In this investigation, however, there appears a fixed set of parameters that are obtained from the reactor and these are kept consistent throughout the input to all three codes (FEVER, EXTERMINATOR, and FLARE) for a particular problem. The input methods for the three codes are not identical, but all required input data does come directly or indirectly from the model reactor core.

A model reactor was chosen for which parameters could easily be obtained. The restriction of being a thermal light water reactor is imposed by the selection of computer codes. For these reasons, the boiling water reactor Dresden I was chosen. It is a large commercial reactor capable of producing 700 MWth (200 MWe) and has been operating since October, 1959. Dresden I has been refueled with slightly different types of fuel elements since its startup. For simplicity only type-1 fuel is considered since the other fuels contain Er_2O_3, the properties (density and cross sections) of which are not available in the existing computer codes.

Table 1 provides the general core characteristics which are of concern.

Table 1

REACTOR CORE DESCRIPTION

DESCRIPTION	VALUE	REFERENCE
Number of assemblies	464	(30)
Number of control rods	80	(30)
Thermal power	700 MW	(30)
Control Rods		
shape	Cruciform	(30)
material	Natural B_4C granules in SS tubes	(30)
	2% boron in steel	(31)
blade span	6.50 inches	(30)
blade thickness	0.375 inches	(30)
poison length	103.0 inches	(30)
Moderator	H_2O	(31)
Coolant	H_2O	(31)
inlet temperature	505°F	(31)
outlet temperature	547°F	(31)
pressure	1015 psia	(31)
Fuel (Type 1)		
rod O.D.	0.567 inch	(30)
pellet O.D.	0.498 inch	(30)
material	UO_2	(30)
active fuel length	106.5 inches	(30)
enrichment	1.5 w/o U-235	(30)
number rods/assembly	36	(30)
temperature (avg. operating)	1200°F	(30)
Cladding		
thickness	0.033 inch	(30)
material	Zr-2	(30)
temperature (avg. operating)	610°F	(30)
Channel		
material	Zr-2	(30)
inside dimension	4.29 inches	(30)
thickness	0.060 inch	(30)
Pressure Vessel (molybdenum-bearing	carbon steel)	(31)
diameter, inside	12.2 ft.	(31)
wall thickness	5.5 inches	(31)
Cell dimension (4 assemblies)	9.962 inches	(30)

A cross-sectional view of the reactor is presented in Figure 1. It is composed of 464 fuel elements or assemblies each of which contain 36 fuel pins or rods. These fuel rods consist of UO_2 fuel in the form of pellets enriched to 1.5 weight percent U^{235}. The cladding material of the pins as well as the assembly walls (open end boxes which house the fuel pins) is Zircaloy-2.

For control there are 80 cruciform control rods which are inserted in the adjoining spaces between assemblies. These rods, Figure 2, are composed of natural B_4C granules in stainless steel tubes. They are inserted from the bottom of the core and only into the innermost fuel assemblies. This inner region which is most directly affected by the control rods is designated as region 1.

Surrounding this region, there are a number of fuel elements whose adjacent gaps cannot be penetrated by control rods. Region 2, as this is called, is identical in composition to region 1 except for the control rods.

Beyond these regions in the radial direction there exists a cylindrical region that contains only light water. This constitutes region 3 and serves as both moderator and coolant (thermal shield). It is not a power generating region.

The outer casing of these three regions is the pressure vessel composed of molybdenum-bearing carbon steel with an internal cladding of stainless steel. This is the fourth region and is approximated

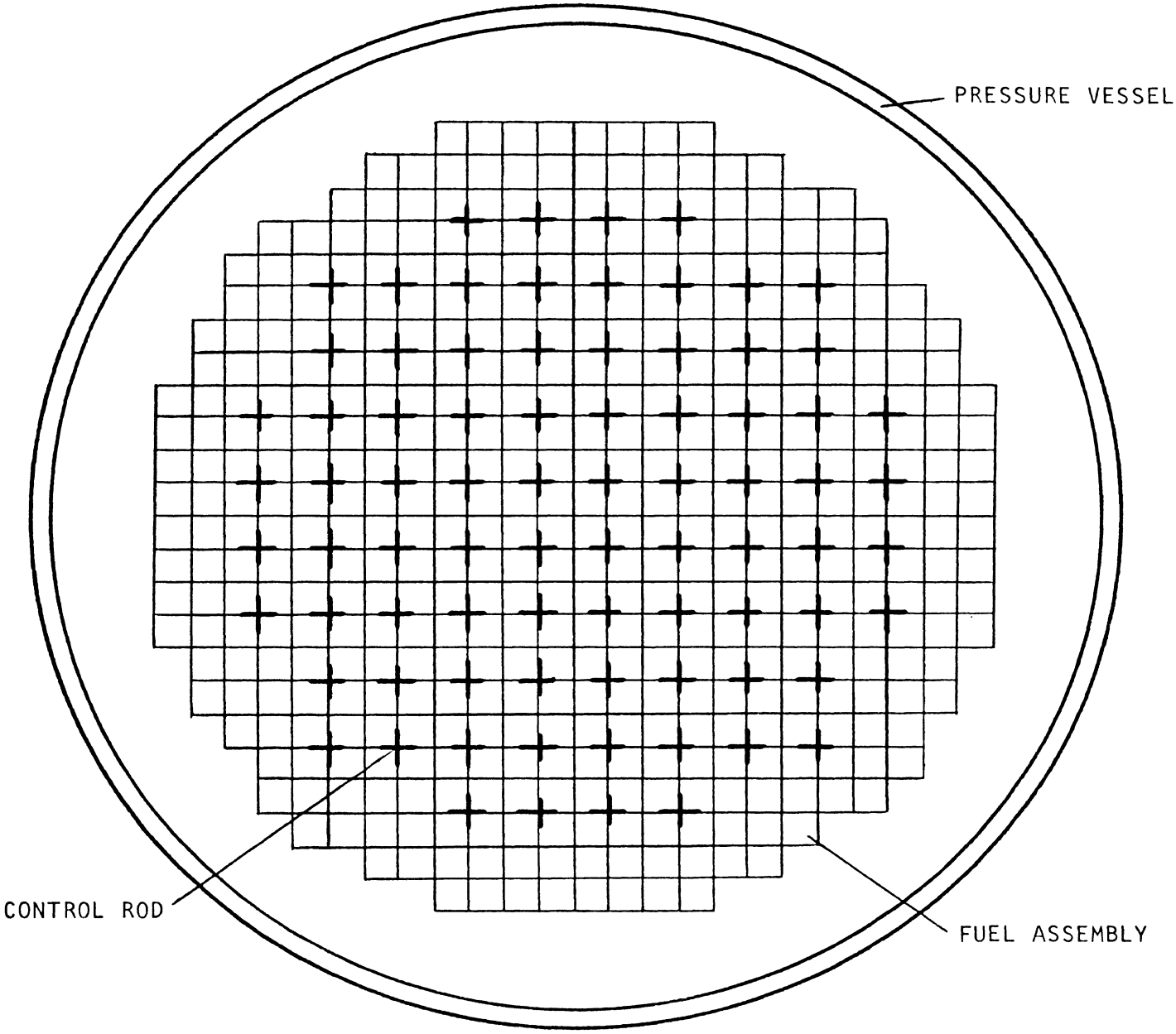


FIGURE 1

CROSS SECTION OF DRESDEN I REACTOR

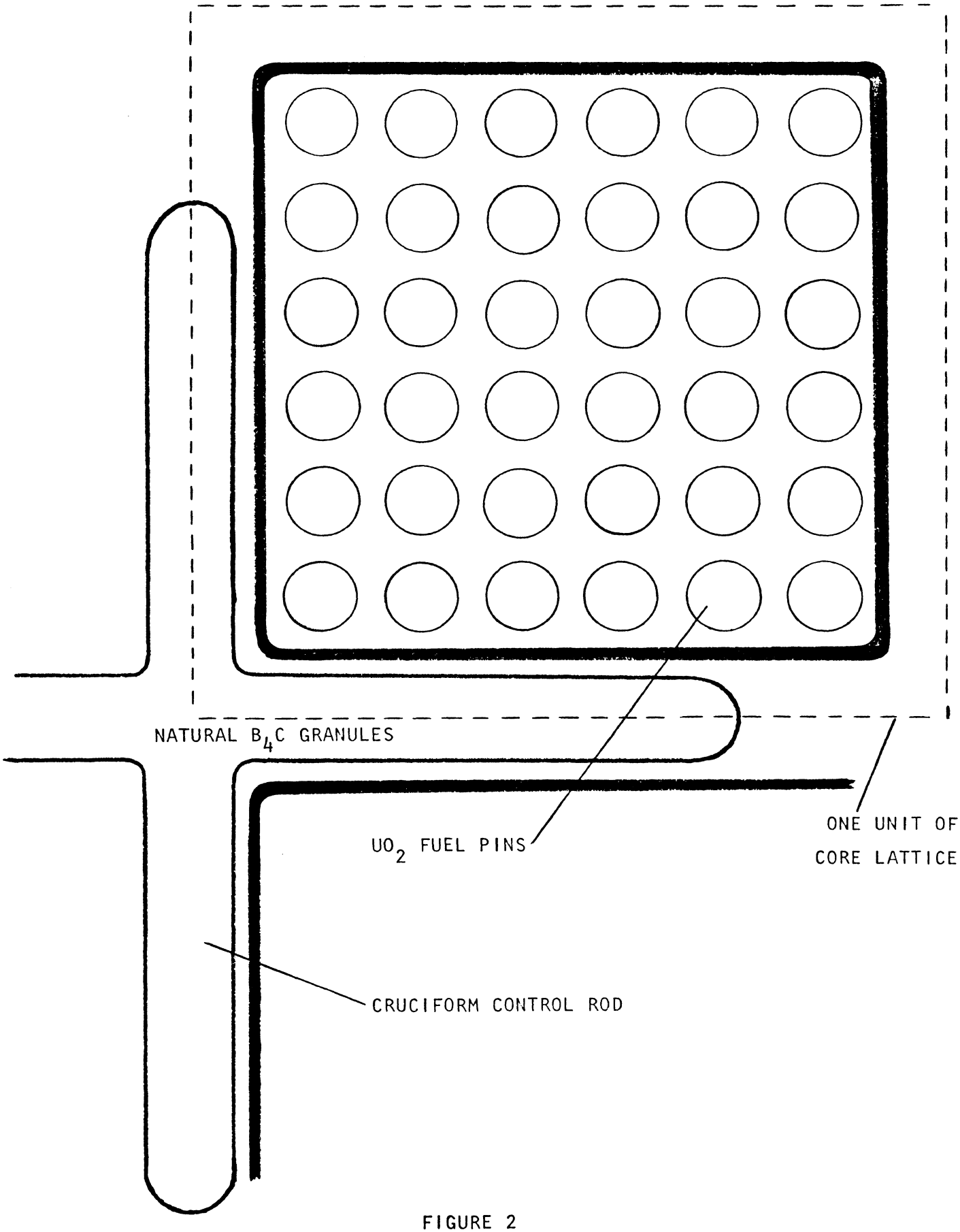


FIGURE 2

DETAILED VIEW OF DRESDEN I CORE LATTICE

as pure iron.

The boundaries of regions 1 and 2 are not smooth surfaces. To allow easier description of the boundaries, these regions are approximated as cylinders. Figure 3 gives the resulting picture of this approximation. The radii of each region must then be determined.

Region 1 is a solid cylinder whose volume is equal to 320 (80x4) units. [A unit is defined as one-fourth of a cell or an assembly with surrounding water. The dashed line in Figure 2 encloses 1 unit. (Unit Area = 24.81 sq. in.)] Its radius is the radius of the equivalent cross sectional area of the innermost 320 units. Therefore,

R_o^2 = (320)(24.81) sq. in.

R_o = 50.271 inches = radius of region 1

The area of region 2 = $\pi(R_1^2 - R_o^2)$ = (24.81)(464 - 320)

R_1 = outer radius of region 2 = 60.534 inches

Inside radius of pressure vessel = 6.1 ft.
Therefore, outer radius of region 3 = 73.2 inches

Pressure vessel wall thickness = 5.5 inches
Outer radius of region 4 = 73.2 + 5.5 = 78.7 inches

Table 2 gives a summary of the region boundaries.

Table 2

REGION BOUNDARIES

Region	Outer Radius (in.)	Outer Radius (cm.)
1	50.271	127.688
2	60.534	153.757
3	73.200	185.928
4	78.700	199.898

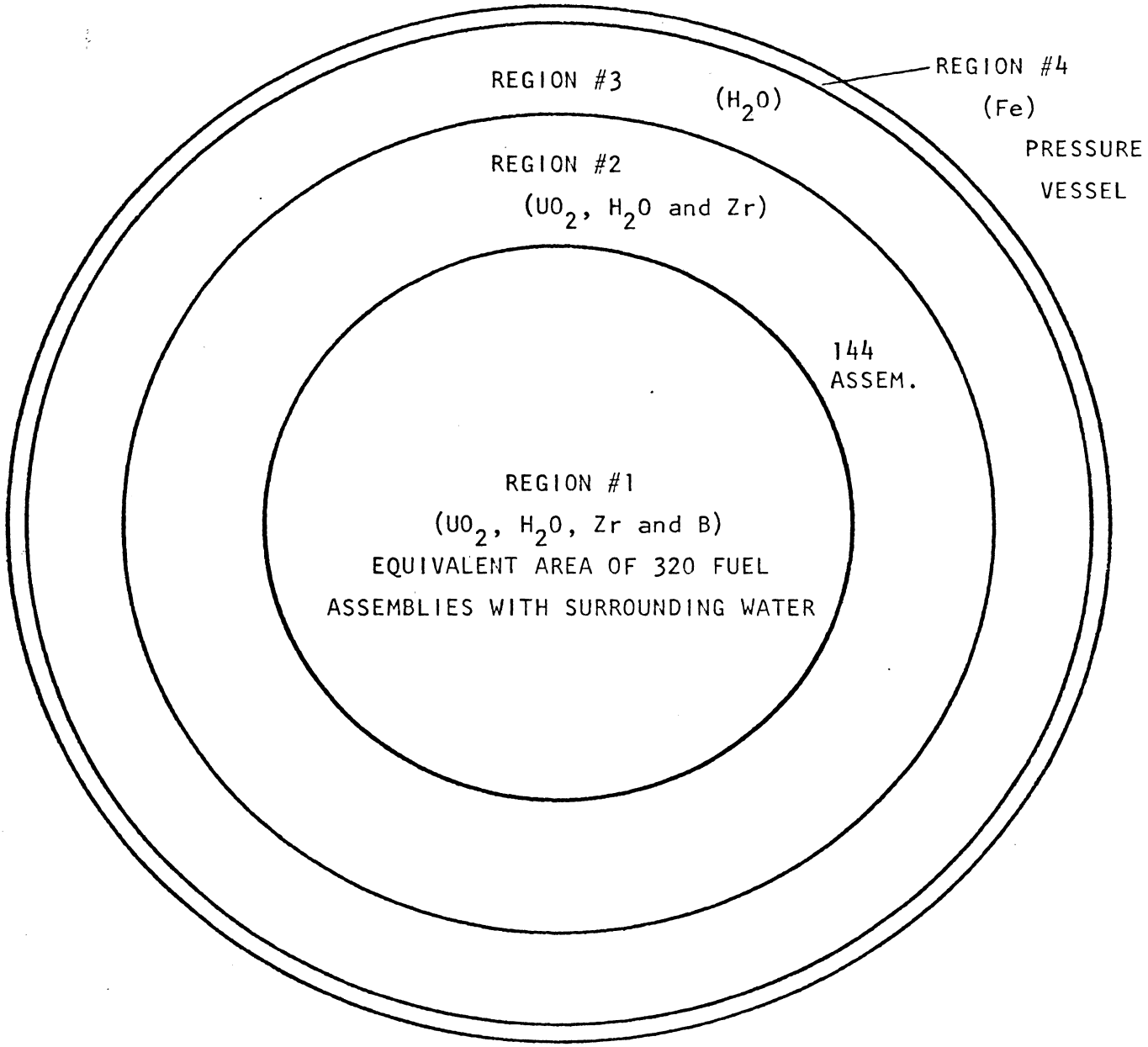


FIGURE 3

APPROXIMATED CYLINDRICAL REGIONS OF DRESDEN I

Besides breaking the reactor into the above four regions, homogenized number densities of the nuclides in each region must be determined. It is assumed that the reactor has not been operated yet and that the core composition is uniform in the axial direction. This simplifies the number density calculations to computing cross-sectional areas of materials in regions one and two (here taken as identical compositions since control poison number densities will be computed separately and assumed to have no effect on other nuclide number densities). The materials involved are the channel walls, fuel, and cladding.

Since the lattice is uniform, one assembly with surrounding H_2O is representative of the fuel regions. This was defined earlier as one unit.

The total area of one unit = $(9.962/2)^2$ = 24.81 sq. in.

Fuel pin cladding area (1 pin) = $\pi(r_1^2 - r_o^2)$ = .05536 sq. in./pin

where r_o = radius to inside surface of clad

r_1 = radius to outside surface of clad

Total area of fuel cladding for one assembly

$$= 0.05536 \frac{\text{sq. in.}}{\text{pin}} \times 36 \frac{\text{pins}}{\text{assem.}} = 1.9930 \text{ sq. in./assem.}$$

Area of channel (assembly wall) = circum. x thickness

= (16.52)(0.060) = 0.9915 sq. in.

Total area of Zr-2 in one unit = 1.9930 + 0.9915 = 2.9845 sq. in.

$$\text{Fraction of Zr-2 in the core} = \frac{\text{Area of Zr-2}}{\text{Area of one unit}} = \frac{2.9845}{24.81} = \boxed{0.12029}$$

The volume fraction of fuel in the core is computed in a similar manner.

Fuel pellet area = πr_p^2 = 0.19478 sq. in.

where r_p = pellet radius

Total area of fuel for one assembly (36 pins) = 7.0122 sq. in.

$$\text{Fraction of fuel in the core} = \frac{\text{Area of fuel}}{\text{Area of one unit}} = \frac{7.0122}{24.81} = \boxed{0.28263}$$

The remainder of the cross-sectional area is H_2O

The final volume fractions for regions 1 and 2 are found to be:

UO_2	0.28263
Zr-2	0.12029
H_2O	0.59708
Total	1.00000

For regions three and four the above calculations are unnecessary since region three is pure water (or steam) and region four is taken as pure iron.

The homogenized number densities are computed using

$$N_{mn} = \frac{V_{mn}\rho_m N_a}{A_m}$$

where

N_{mn} = number of molecules or atoms/cm^3 in region n for element or compound m.

V_{mn} = volume fraction of element or compound m in region n.

ρ_m = density of nuclide or compound m.

N_a = Avagadro's number = 0.6023×10^{24}.

A_m = atomic weight of element or compound m.

As an example, the homogenized number density of Zr is illustrated below.

$$\frac{\#Zr \text{ atoms}}{cm^3 \text{ of region 1}} = N_{Zr,1} = \frac{V_{Zr,1} \cdot \rho_{Zr} \cdot N_a}{A_{Zr}}$$

$$= \frac{(0.12029)(6.49)^{**}\ (0.6023 \times 10^{24})}{91.22^{**}} = 0.0051546 \ \frac{\text{atoms}}{\text{barn-cm}}$$

The remaining number densities are determined in similar fashion. U^{235} and U^{238} are handled on a precentage basis (1.5% U^{235}) of uranium atoms. It must be observed that oxygen is contained not only in the fuel molecules, but also in the water molecules.

A summary of the number densities of the nuclides in the reactor at room temperature are given in Table 3. It is assumed that initially the entire coolant volume is composed of water at 68°F.

Table 3

INITIAL NUMBER DENSITIES† BEFORE STARTUP - 68°F

NUCLIDE	REGIONS 1&2	REGION 3	REGION 4
U^{235}	0.0000964630*	0.00	0.00
U^{238}	0.00633438	0.00	0.00
H	0.0399579	0.0669222	0.00
O	0.0328406	0.0334611	0.00
Zr	0.00515462	0.00	0.00
Fe	0.00	0.00	0.0847642

* per barn-cm

† Six-digit accuracy for computational purposes only; last three digits are usually not meaningful.

** From Reference (32)

As the reactor is operated, the coolant temperature is raised to approximately 500°F. This causes a density change in the moderator and hence in the reactivity. Only the water undergoes an appreciable change in density thus affecting the number densities of hydrogen and oxygen in regions 1, 2, and 3. Table 4 gives the number densities for a water density at operating temperature and pressure.

Table 4

NUMBER DENSITIES AT OPERATING TEMPERATURES - ENTIRE COOLANT VOLUME COMPOSED OF WATER @ 500°F. $\rho_w^* = 0.785 g/cm^3$

NUCLIDE	REGIONS 1&2	REGION 3
H	0.031366*	0.0525326
O	0.028545	0.0262663

Other nuclide number densities remain the same.

As the Dresden I reactor is operated, boiling increases, reducing the water level. The remainder of the core is occupied by saturated steam of ≃ 5% quality. The following calculation determined the density of this steam at 545°F and 1015 psia.

$$\text{quality} = \frac{M_g}{M_g + M_f} = 5\% = 0.05$$

$$v = v_f + \frac{M_g}{M_g + M_f}(v_g - v_f)^{**} = 0.0216 + (.05)(.4456-.0216)^{\dagger} = 0.0428 \frac{ft^3}{lb.}$$

$$\rho = \frac{1}{v} = 0.374065 \text{ gm/cm}^3$$

* From Reference (33)

** From Reference (34)

† From Reference (35)

where v = specific volume

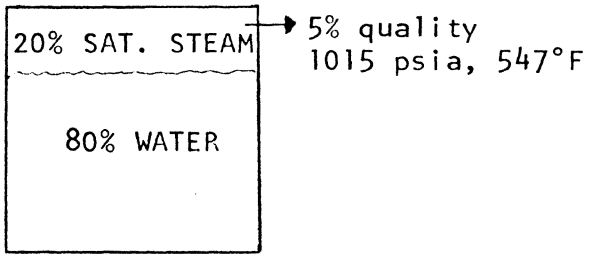
v_g = specific volume of saturated vapor

v_f = specific volume of saturated liquid

M_g = mass of gas

M_f = mass of fluid

Using the above calculated density of saturated steam in the void space, the number densities can be found for any combination of water and steam that may occur in the reactor. The mixture of 20% steam and 80% water[36] is found to be a reasonable estimate shortly after startup. It remains with this proportion throughout the remainder of the operation.



Presuming the density of the structural materials and the fuel remain constant, the hot water and saturated steam number densities are weighted by their respective volume fractions. Table 5 summarizes these values.

Table 5

NUMBER DENSITIES AT OPERATING TEMPERATURES AND 1015 PSIA

NUCLIDE	REGIONS 1&2	REGION 3	-VOLUME FRACTIONS OF COOLANT- HOT WATER	SAT. STEAM
H	.0280820*	.0470326	0.80	0.20
O	.0269027	.0235164	0.80	0.20

* per barn-cm

Other nuclide number densities remain the same.

B. INPUT PREPARATION FOR CODES

1. Cross Section Calculations

In order to solve the neutron diffusion or transport equations, a detailed knowledge of the neutron microscopic cross sections for each material in the reactor must be known for various processes. Accuracy in design studies is dependent upon the reliability of these cross sections. However, there still remains the error introduced by working with an energy spectrum of discrete groups rather than one which is continuous. Many times the number of groups is arbitrary within a specified limit for a particular code. The complexity of the problem and the computer time consumed increase in proportion to the number of groups specified.

The number of energy groups for this investigation was set at four: one thermal, one epithermal, and two fast. This is the maximum number allowed by the FEVER code and hence is kept constant for EXTERMINATOR-2. The three-dimensional code FLARE uses only one group of neutrons.

The nuclides which are of concern are those contained in the structural materials, fuel, coolant and moderator, control rods, fission chains, and fission products. Specifically they are U^{235}, U^{236}, U^{238}, Np^{239}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}, fission product, Xe^{135}, Sm^{149}, B, H, O, Zr, and Fe. For each nuclide the group dependent absorption, fission, transport, and scattering cross sections must be obtained. The average number of neutrons per fission must also be known.

The energy spectrum must be broken down into distinct energy groups. The codes described use the following groups.

BROAD GROUP	ENERGY RANGE (37)
1	10 MEV → 67.4 kev
2	67.4 kev → 1.23 kev
3	1.23 kev → 0.683 kev
4	0.683 ev → 0.0

For the fast group constants, the HRG code was used. It is similar to GAM-1 and has a more extensive library of cross sections for a wider variety of nuclides. It also includes resonance absorption in more isotopes than U^{238} and Th^{232}.

The HRG code solves the time independent Boltzmann equation with isotropic sources.

$$\vec{\Omega}.\nabla N(\vec{r},E,\vec{\Omega}) + \Sigma_T N(\vec{r},E,\vec{\Omega}) = \frac{S(\vec{r},E)}{4\pi} + \iint \Sigma_s (E'\to E,\vec{\Omega}'\to\vec{\Omega}) N(E',\vec{r},\vec{\Omega}')\, dE'\, d\Omega'$$

where $N(\vec{r},E,\vec{\Omega})$ is the number of neutrons with energy E crossing a unit surface at $\vec{r}$ per unit time going in a unit solid angle centered in the direction $\vec{\Omega}$.

The fluxes and current terms are calculated using the P-1 approximation at each of the 68 groups in its library. It also uses the previously calculated number densities and the reactor temperatures for moderator, cladding, and fuel. Number densities of nuclides which do not occur initially in the core are input as a small number (1% of U^{235}). This causes negligible effect on the neutron energy spectrum but allows printout of microscopic cross

sections. The option to obtain microscopic cross sections was used for each isotope averaged over each broad group. The program also supplied the macroscopic broad group cross sections, diffusion coefficients, age, and group transfer coefficients.

The third broad energy group (epithermal) is unlike the first two in the respect that certain nuclides exhibit pronounced resonance cross sections in this energy range. Normally, resonance absorption is accounted for by the narrow resonance or narrow resonance infinite mass calculations of the resonance integral. The method used by GAM-1 and HRG is essentially that of Adler.[38] The values that are needed as input are

σ_m = moderator cross section per absorber atom(infinite mass approximation)

σ_p = total scattering cross section per absorber atom (narrow resonance)

$\sigma_m(eff)$ = effective cross section for unresolved resonance calculations

$$\sigma_m = \frac{\Sigma_S^M}{N}$$

Σ_S^M = macroscopic non-resonance scattering cross section for all nuclides in the fuel lump other than the resonance nuclide being considered.

N = number density of the resonance nuclide (in fuel).

The isotopes considered for resonance properties are U^{235}, U^{238}, Pu^{239}, Pu^{240}, and Pu^{241}. The plutonium number densities are assumed to be 1% of N^{U235}. Therefore

$$\Sigma_S^M = (N\sigma_s)^{U\text{-}235} + (N\sigma_s)^{O\text{-}16} + (N\sigma_s)^{Pu\text{-}239} + (N\sigma_s)^{Pu\text{-}240} + (N\sigma_s)^{Pu\text{-}241}$$

The microscopic scattering cross sections were obtained from BNL-325[39] at approximately 1 Kev.

$\sigma_p = \frac{\Sigma_p}{N}$ where Σ_p is the total macroscopic potential scattering cross section including the resonance nuclide itself. This can be simplified to a more workable equation for a particular nuclide.

$$\sigma_p^{NUC} = \frac{(N\sigma_s)^{NUC} + \sum_{i \neq NUC} (N\sigma_s)^i}{N^{NUC}} = \sigma_s^{NUC} + \sigma_m^{NUC}$$

$$\sigma_m(\text{eff}) = \sigma_p + \sigma_m \Big/ (1+\sigma_m/\sigma_\ell)$$

where $\sigma_\ell = 1/N\bar{\ell}_o$ and $\bar{\ell}_o$ = mean chord length in the absorber lump

$\simeq$ diameter of the fuel

= 1.265 cm

All number densities are for pure fuel lump.

To better illustrate this procedure, the case for U^{235} is worked through below.

(1.5%) $N^{U235} = 0.0003413 \times 10^{24} \frac{\text{atoms}}{\text{cm}^3 \text{ of fuel}}$

Σ_S^M = (all negligible except U^{238} and oxygen)

$= (N^{U238})(\sigma_s^{U238}) + (N^{O})(\sigma_s^{O}) = 0.37464 \text{cm}^{-1}$

$$\sigma_m^{U235} = \frac{\Sigma_S^M}{N^{U235}} = \frac{0.37464}{0.0003413} \text{ barns} = 1097.6 \text{ barns}$$

$\sigma_s^{U235} \simeq 15.0$ barns

so $\sigma_p^{U235} = \sigma_s^{U235} + \sigma_m^{U235} = 15.0 + 1097.6 = 1112.6$ barns

$\sigma_\ell^{U235} = 1 / (N^{U235} \bar{\ell}_o) = 2316.4$ barns

$\sigma_m^{U235}(\text{eff}) = \sigma_p^{U235} + \sigma_m^{U235} / (1+\sigma_m^{U235} / \sigma_\ell^{U235}) = 1857.4$ barns

The mean chord length $\bar{\ell}$ must also be calculated for input. It is determined from

$$\bar{\ell} = \bar{\ell}_o \left[\frac{\lambda + (4v_1/S_o)}{4v_1/S_o} \right]$$

where λ = mean free path in the moderator

v_1 = moderator volume

S_o = absorber surface

Table 6 gives the results of these input parameters.

Table 6

NUCLIDE	HRG RESONANCE ABSORPTION VALUES σ_m(barns)	σ_p(barns)	σ_m(eff)(barns)	Lumped nuclide density (per barn-cm)
U^{235}	1097.6	1112.6	1857.34	3.4130×10^{-4}
U^{238}	7.945	16.945	23.430	2.2412×10^{-2}
Pu^{239}	1.113×10^5	1.113×10^5	1.8648×10^5	3.4×10^{-6}
Pu^{240}	1.113×10^5	1.113×10^5	1.8648×10^5	3.4×10^{-6}
Pu^{241}	1.113×10^5	1.113×10^5	1.8648×10^5	3.4×10^{-6}

$\bar{\ell}$ (cold) = 1.3644 cm.

$\bar{\ell}$ (hot operating) = 1.5308 cm.

The thermal neutron group cross sections are obtained using the TEMPEST II code. An infinite lattice is assumed and its composition is identical to that of regions one and two of the reactor core. For this investigation the thermal neutron flux spectrum was based upon the Wigner-Wilkins light moderator equation. Microscopic cross section averages over that spectrum were then obtained.

For the initial nuclides in the lattice, the homogenized number densities are required. The temperature of the system must also be known as well as the maximum and minimum energy

limits for each group. A buckling is also required but may be estimated if not precisely known.

The output from TEMPEST II includes the total flux and average diffusion coefficient in addition to microscopic absorption and fission cross sections computed from

$$\overline{\sigma_a} = \frac{1}{\Phi} \int_{E_{min}}^{E_{max}} \phi\ (E)\ \sigma_a\ (E)dE$$

$$\overline{\sigma_f} = \frac{1}{\Phi} \int_{E_{min}}^{E_{max}} \phi\ (E)\ \sigma_f\ (E)dE$$

where $\Phi = \int_{E_{min}}^{E_{max}} \phi\ (E)dE$ $\qquad$ $E_{max} = 0.683$ ev

$E_{min} = 0.0$

The quantity

$$\overline{(1-\bar{\mu})\sigma_s} = \frac{1}{\Phi} \int_{E_{min}}^{E_{max}} \phi\ (E)(1-\bar{\mu})\ \sigma_s\ (E)dE$$

is also computed aiding in the determination of the microscopic transport cross sections. No downscatter cross sections are needed since there is no lower energy group than thermal.

Since not every nuclide is available on the TEMPEST library, a few cross section values were obtained from either the library of the FEVER[3] report sample problem or by running the LASER[9] code. Table 7 lists the results of the thermal cross section calculations for the reactor at (1) room temperature, (2) operating temperature-full core water, and (3) operating temperature-full core saturated

TABLE 7 THERMAL ENERGY GROUP CONSTANTS FOR COLD (68°F) CORE

NUCLIDE	$\overline{\sigma_a}$	$\overline{\sigma_{tr}} = \overline{\sigma_a} + \overline{\sigma_s(1-\bar{\mu})}$	$\overline{\sigma_f}$	ν
U235	0.5055E+03	0.5155E+03	0.4302E+03	2.43
U236	0.5369E+01	0.1534E+02		
U238	0.2078E+01	0.1015E+02		
Np239	0.2272E+02*	0.3269E+02*		
Pu239	0.1032E+04	0.1042E+04	0.7063E+03	2.91
Pu240	0.2302E+03	0.2416E+03	0.2127E-01	2.89+
Pu241	0.1018E+04	0.1028E+04	0.8510E+03	3.06
Pu242	0.1013E+02*	0.1970E+02*		
F.P.	0.2114E+01*	0.1011E+02*		
Xe135	0.2765E+07	0.2765E+07		
Sm149	0.8610E+04	0.8618E+04		
B	0.3127E+03*	0.3165E+03*		
H	0.2545E+00	0.3071E+02		
O	0.1542E-03	0.4025E+01		
Zr	0.1418E+00	0.6396E+01		
Fe	0.2008E+01	0.1288E+02		

* from FEVER report test prob. library

+ from LASER calculation

steam. Only the cold core thermal cross sections are illustrated. The other two cases are similar.

By having the three cases available, it is possible to formulate the following problems:

(1) Cold clean reactor
(2) Hot operating - no steam voiding
(3) Hot operating - any precentage of steam voiding

Appendix I gives a listing of a computer program developed to volume weight the hot water and saturated steam cross section blocks.

With the above cross sections at hand and the previously defined core as reference, any input requirements that are needed for FEVER, EXTERMINATOR-2, and FLARE can be satisfied.

2. One-Dimensional FEVER Code

The one-dimensional code FEVER[3] was developed primarily to evaluate the effects of fuel depletion and control rod movement in an HTGR. However, due to its generality, it can be used for most types of reactors including light water BWR's.

The multigroup multiregion diffusion equations are solved using a maximum of four groups and twenty regions. It limits slowing down of neutrons to only the next lower energy group. Therefore, scattering is limited to $\sigma_{g,g+1}$ where g represents a particular group.

The diffusion equations using cylindrical geometry result in

$$\text{(fast group \#1)}\quad -D_1\frac{1}{r}\frac{\partial}{\partial r}\left(r\frac{\partial\phi 1}{\partial r}\right) + (\Sigma_{a1}+\Sigma_{s1}+D_1B_1^2)\phi_1 = \frac{1}{k_{eff}}\sum_{i=1}^{4}\nu_i\Sigma_{f_i}\phi_i$$

$$\text{and}\quad -D_i\frac{1}{r}\frac{\partial}{\partial r}\left(r\frac{\partial\phi i}{\partial r}\right) + (\Sigma_{ai}+\Sigma_{si}+D_iB_i^2)\phi_i = \Sigma_{s_{i-1}}\phi_{i-1} \qquad \text{for} \quad 2\leq i\leq 4$$

where the radial distance (r) is the independent variable. A two-dimensional synthesis can be made by specifying a buckling B_Z^2. This will approximate the leakage in the axial direction. All neutrons are born in the highest energy group, the source for the remaining groups being the slowing down from the above group. The diffusion calculation is completed when convergence on k_{eff} and the flux is reached.

The burnup calculations utilize the two heavy-isotope chains. The relationships

$$\frac{dN^{U238}}{dt} = -N^{U238}\,\sigma_a^{U238}\,\phi \qquad \underline{\text{for } U^{238}}$$

$$\frac{dN^{Np239}}{dt} = N^{U238}(\sigma_a-\sigma_f)^{U238}\,\phi - N^{Np239}(\lambda^{Np239} + \sigma_a^{Np239}\,\phi) \qquad \underline{\text{for } Np^{239}}$$

etc.

are used to determine the burnup and production of the nuclides as a function of time. Forward differences are used to solve the first order differential equations.

Input to FEVER includes:

(1) cross section blocks which remain constant throughout the problem

(2) initial concentrations per region for each nuclide

(3) region radii and core height

(4) burnup time step, thermal power of reactor, and fission rate

Since the control-poison search is used, the boron number density is adjusted at each time step to allow the reactor to remain critical. If the control poison is completely absent and $k_{eff} < 1.0$, the problem will terminate.

The output contains:

(1) control poison number densities for the critical case

(2) maximum k_{eff} at operating temperature with all control poison removed

(3) k_{eff} at cold shutdown with xenon removed and all control poison fully inserted

(4) all nuclide concentrations by region

(5) conversion ratio

(6) weight of heavy isotopes

(7) flux and power distributions

Fever was modified at UMR in August, 1968 to allow it to run on the IBM 360/50. The previous deck was obtained from Purdue University in March, 1968.

3. Two-Dimensional EXTERMINATOR-2 Code

The multigroup multiregion EXTERMINATOR-2[13] code was employed to perform the two-dimensional computations. It is capable of solving the neutron diffusion equations similar in nature to those found in FEVER for XY, RZ, or RΘ geometry. RZ was chosen since it yields the most useful information for the model core.

The multigroup neutron diffusion equations which are approximated by difference equations are solved by the Equipoise method.

Normally, numerical iteration methods for solving the diffusion equations result in the use of an inner iteration (flux) and an outer iteration (source). When one type of iteration has met a convergence criterion, the other type of iteration is used and vice versa. Convergence in both at the same time determines completion. The Equipoise scheme is founded on the idea that separate inner and outer iterations are not needed and merging of the two processes is beneficial. The basic feature of the method is the use of the most recently computed flux values in the iteration process.

Since the Dresden I reactor is assumed to be symmetric in both the axial and radial directions, only a quarter core was analyzed. It was, therefore, necessary to assume symmetry boundaries on the bottom and left edges and zero flux at the top and right sides. Figure 4 gives a view of the cross section and mesh spacing used. Twenty-five columns of mesh points are taken in the radial direction and ten rows of mesh points in the axial direction. Note that the actual center line of the reactor is located between columns 1 and 2 and that the bottom border is situated midway between rows 9 and 10. The zero flux boundaries coincide with row 1 and column 25.

The microscopic cross sections used in FEVER are also used in the EXTERMINATOR code. The computer program described in Appendix II simplifies this conversion. EXTERMINATOR allows for upscatter and downscatter to any other group whereas FEVER allows only downscatter to the next lower group. Despite the added

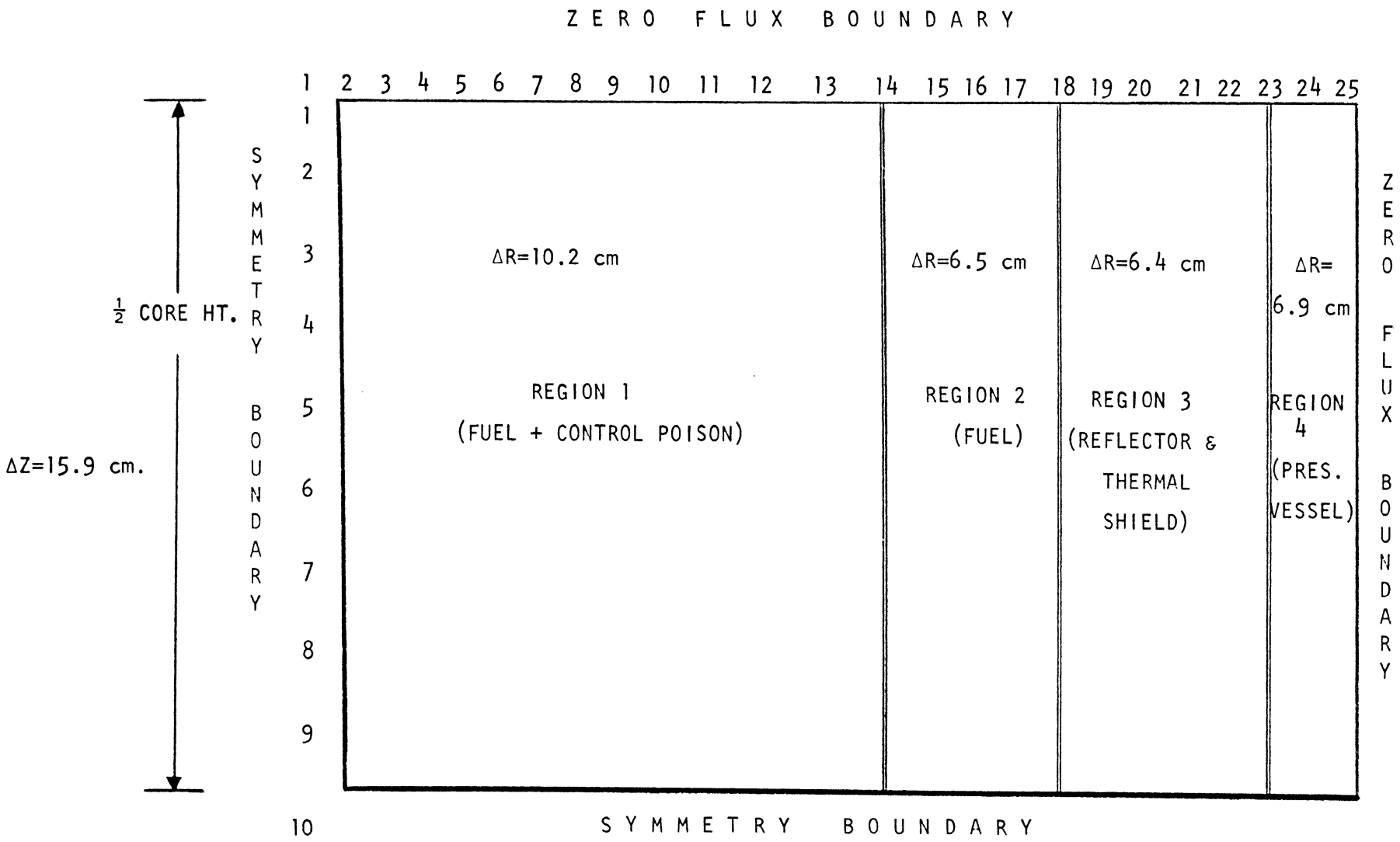


FIGURE 4

RECTANGULAR REGION OF SOLUTION IN R-Z PLANE (EXTERMINATOR GRID)

flexibility, only scattering to the next lower group is used in EXTERMINATOR for two reasons: (1) to keep the input as consistent as possible between programs, and (2) the upscatter and remaining downscatter cross sections are not available from HRG and TEMPEST.

Burnup is not accounted for in EXTERMINATOR since it is merely a diffusion code. However, by relying on current number densities from FEVER at each time step, k_{eff}, flux, etc. can be obtained for the two-dimensional case. FEVER prints out region dependent nuclide densities at each time step.

The basic input required by EXTERMINATOR includes:

(1) number of groups and compositions
(2) type of geometry and boundary conditions
(3) microscopic cross sections
(4) mesh spacings and core dimensions
(5) concentrations of nuclides

The output exclusive of options contains:

(1) diffusion constants by group and region
(2) scattering matrix
(3) convergence data and k_{eff}
(4) reaction rates for each nuclide
(5) relative fluxes for each group at each mesh point
(6) group neutron balance over all compositions

4. Three-Dimensional FLARE Code

The three-dimensional calculations were performed in a manner unlike those in one and two dimensions. The FLARE[22] code

uses a modified one-group diffusion theory which involves only the infinite multiplication factor and migration area M^2. In addition to determining the core reactivity and power distribution, allowance is made for independent movement of each control rod. By working in three dimensions, a coarser mesh must be tolerated. The maximum grid permitted on the UMR version is 17x17x10 in the X, Y, and Z directions respectively. For a quarter core of Dresden I (see Figure 5), $X_{max} = 12$, $Y_{max} = 12$, and $Z_{max} = 10$. These values were chosen specifically to allow each mesh point in the XY plane to coincide with the center of each fuel element. To permit more detail and still conserve computer time and storage, only fuel regions are considered. Compensation is made for the reflector by replacing it with appropriate albedos at the core-reflector interface. Large reactors may be considered by using the quarter core option with symmetry boundaries or by considering the quarter core as a full core and adjusting the albedos interior to the core. Figure 5 gives a clear description of the quarter core and albedos used.

FLARE utilizes a three level scheme for its iteration process. The three levels are (1) source or power iteration, (2) void iteration, and (3) fuel burnup iteration. The basic source iteration equation is

$$S_\ell = \frac{k_\infty \Sigma' S_m W_{m\ell}}{1 - k_{\infty_\ell} W_{\ell\ell}}$$ where S_ℓ is the rate of production of fission energy neutrons at node (cube of core volume) ℓ. $W_{m\ell}$ is the transport kernel (probability that a neutron born at node ℓ is

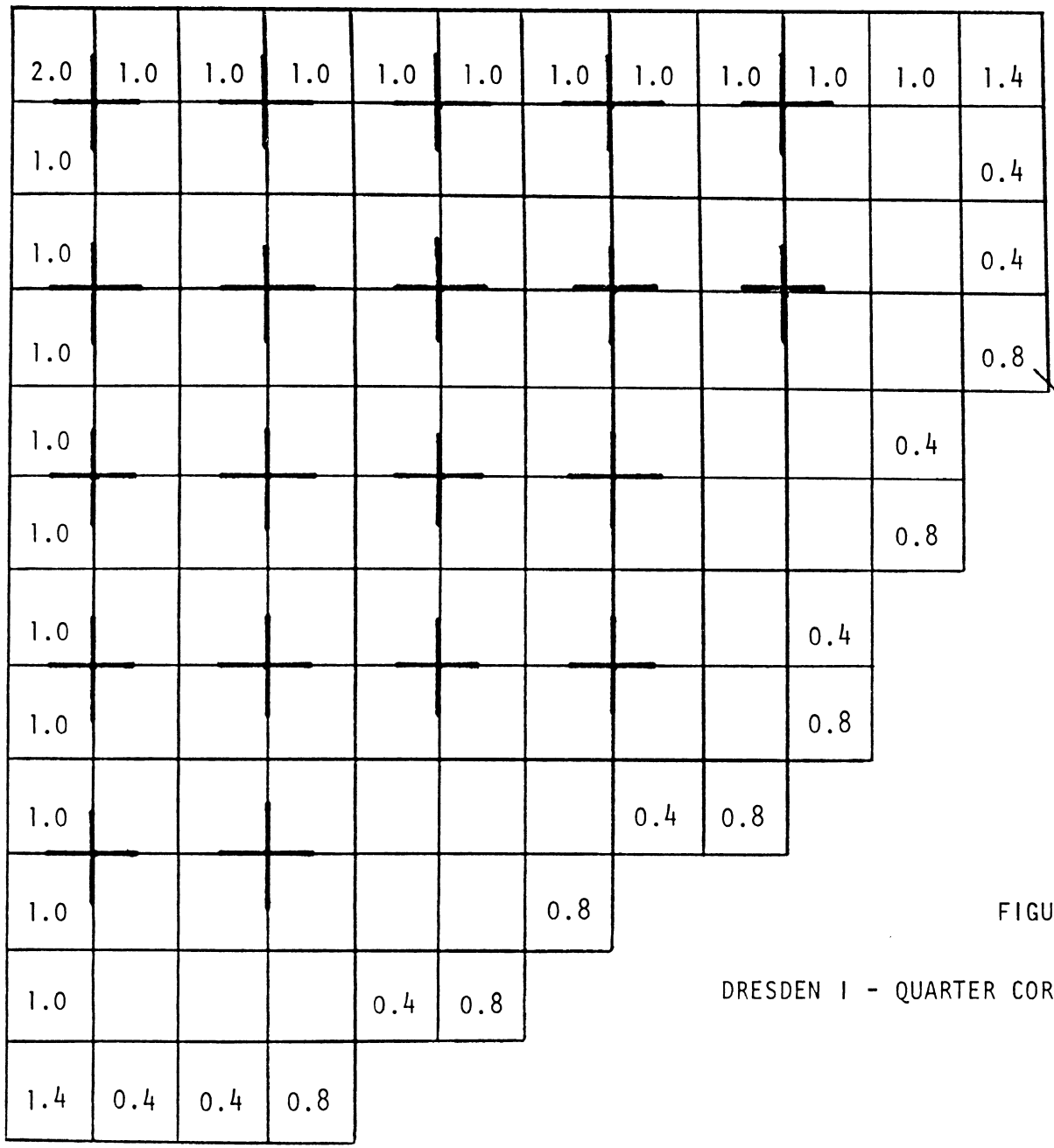


FIGURE 5

DRESDEN I - QUARTER CORE X-Y PLANE CONFIGURATION

absorbed at node m, and the prime signifies summing over the six adjacent nodes. This will be repeated until convergence or the specified maximum number of iterations is reached.

The void iteration is based on a fitting of steam volume fractions to steam quality. The quality at each node is a function of the inlet and exit quality of steam, power, and coolant channel flow. The latter quantity depends on the power of each channel; the dependence is calculated outside of FLARE and input by means of supplying coefficients for the fitting equation. Steam volume fractions $R_{g_{ijk}}$ for each node depend on quality Q_{ijk} by

$$R_{g_{ijk}} = C_1+C_2Q_{ijk} + C_3(Q_{ijk})^2 - C_4 \exp\left(\frac{C_5-Q_{ijk}}{C_6}\right)$$

where the coefficients C_1 - C_6 are input after performing calculations exterior to the code.

Relative moderator density is determined by $U_{ijk}=1-R_{g_{ijk}}(1-\rho_s/\rho_w)$ where ρ_s and ρ_w are densities of the steam and water respectively.

The fuel burnup iteration determines a new exposure E_{ijk} for each node by

$$E_{ijk} = E_{ijk} + B_{22}\,\Delta E \cdot S_{ijk}$$

where B_{22} is an input coefficient and S_{ijk} is the source strength at mode ijk. With this new exposure, a new multiplication factor, voiding, power distribution, etc., are then calculated.

The basic input to FLARE includes:

(1) thermal and rated reactor power

(2) grid space increments and albedos

(3) coolant flow and subcooling of inlet coolant

(4) B_1-B_3 coefficients used to determine migration area as a function of moderator density

(5) B_4-B_{12} constants relating k_∞ to moderator density and control

(6) B_{13}-B_{23}, B_{26} coefficients which are used in determination of k_∞ and Δk

(7) B_{24} and B_{25} connect the dependence of power and flow rate for each fuel bundle

(8) void coefficients C_1-C_9

(9) control rod positions

The output of interest is:

(1) microscopic and macroscopic k_{eff}

(2) two and three dimensional normalized source values, moderator densities, void fractions, and exposures

IV. DATA AND RESULTS

A. CONVERGENCE AND ASSUMPTIONS

The operations employed to obtain the comparative parameters are displayed in Figure 6. This gives an overall view of each code's importance. In the flow chart, the dashed line connecting the MASS BALANCE of FEVER to the BURNUP calculation of EXTERMINATOR indicates that the isotope masses are input from FEVER into EXTERMINATOR for each time step. This is necessary because EXTERMINATOR is not a burnup code. After the lifetime and the control studies had been completed, a brief power flattening study using three rod patterns was made using FLARE. Only FLARE allows independent movement of control rods.

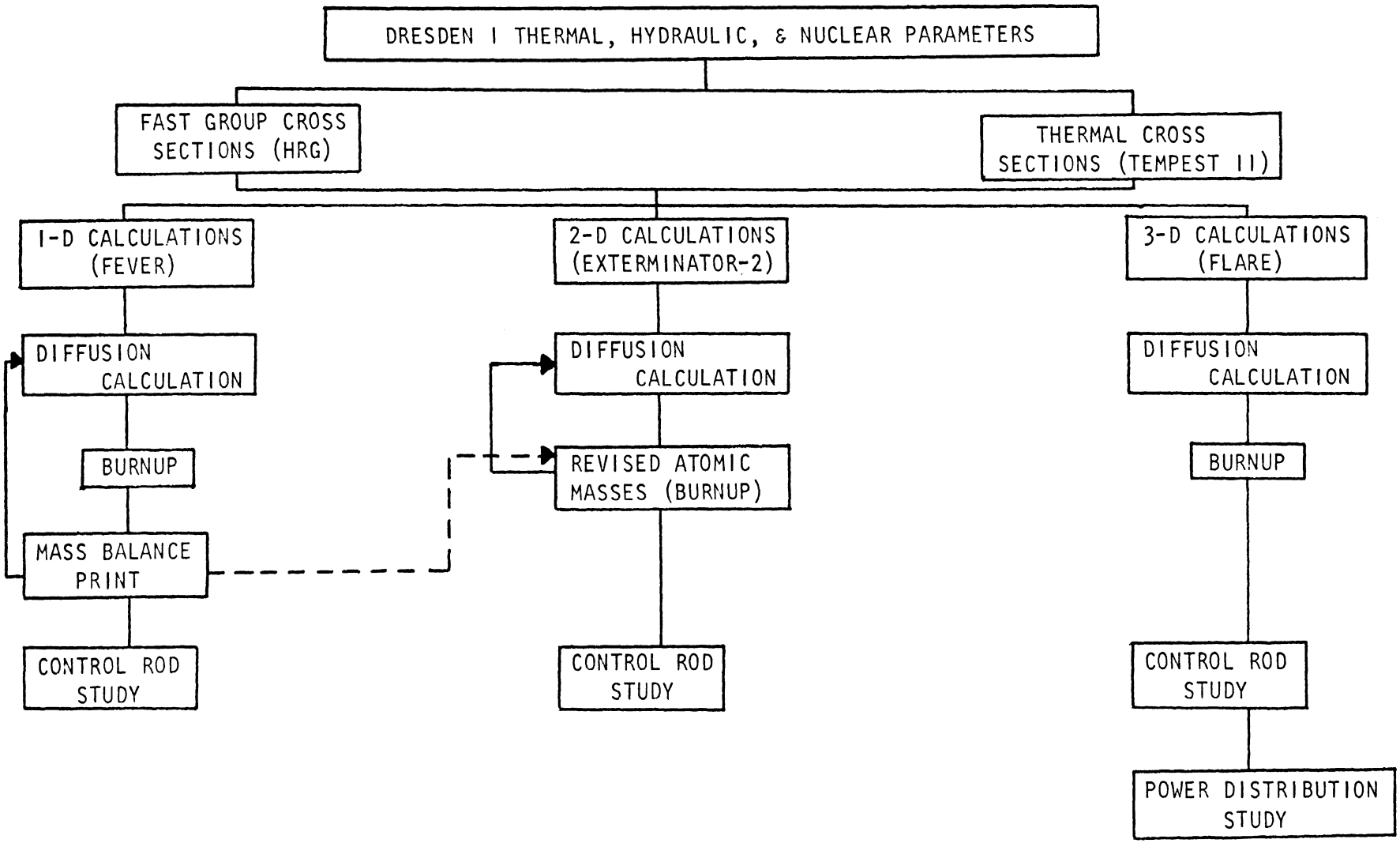
The rate and amount of convergence required are important since all the diffusion codes are iterative in nature. Criteria must be specified for k_{eff} and flux convergence whereas FLARE requires an additional void convergence criterion. Table 8 gives the values used in this work.

TABLE 8 CONVERGENCE CRITERIA

	k_{eff}	Flux	Void
FEVER	10^{-5}	10^{-5}	--
EXTERMINATOR	10^{-4}	10^{-4}	--
FLARE (all)	10^{-3}	10^{-3}	10^{-3}
FLARE*	$.5\times10^{-4}$	$.5\times10^{-4}$	$.2\times10^{-4}$

* Commonwealth Edison FLARE values for burnup steps 0,1, and 2

FIGURE 6 FLOW CHART OF OPERATIONS



Convergence for k_{eff} is much faster than for the neutron flux. The primary reason is that k_{eff} represents a total or macroscopic effect whereas the flux convergence is dependent on microscopic or local conditions. Since the flux at each mesh point is checked, convergence at all points simultaneously may be difficult. This was found to be true for control rod cases especially for EXTERMINATOR and FLARE. For EXTERMINATOR the amount of time needed to reach convergence on the flux was about ten (10) timeslonger than that required for k_{eff} convergence. FLARE demonstrated less than a 0.01% change in k_{eff} for cases differing by a factor of ten in computer time. Therefore, when only a macroscopic value such as k_{eff} is desired, flux convergence need not be overrestrictive.

The elapsed computer times for each particular case or combination of cases is tabulated for the UMR IBM 360/50.

The fortran source decks were compiled and put on disk and/or made into machine language decks. The fact that FEVER and EXTERMINATOR were run from disk whereas FLARE was run as an object deck must be noted when comparing their elapsed computer times.

When cases were stacked one behind another, overall CPU times were observed to decrease. This acceleration in convergence was gained only for cases which were similar in nature and the k_{eff} and neutron flux values were saved from the previous run.

Several assumptions and approximations were made in determining

the cross sections and setting up the reactor model. These assumptions are listed below in two groups of importance.

Assumptions of lesser importance include:

(1) Reactor regions approximated as cylinders

(2) Fuel regions were homogenized

(3) Cross sections were region independent

(4) Fourth region was taken as pure iron (Reactor Vessel)

(5) Cross sections were volume weighted to account for voiding

(6) Some cross sections were estimated when they could not be determined explicitly.

Assumptions of greater importance include:

(1) Cross sections were not time dependent. This is important in lifetime studies where the energy spectrum changes with time hence affecting the flux weighted cross sections.

(2) No thermal disadvantage factor was used since TEMPEST does not account for the thermal flux depression in the fuel.

The cold clean cases for FEVER, EXTERMINATOR, and FLARE with all rods inserted gave k_{eff} values well below 1.000. This is an important safety criterion that must be included in every feasible power reactor design.

B. EXCESS REACTIVITY OVER LIFETIME

Assuming one core of uniform enrichment with no burnup, the excess k_{eff} was determined for the cold clean case (-3)*, the hot-no fission product buildup nor voiding case (-2)*, and operating condition with voiding cases (-1 to 10)*. In the cold case (-3)*, the k_{eff} is the highest. This is due not only to the fresh fuel but also to the greater density of the moderator. The water density will decrease with temperature and hence loose some of its ability to slow down neutrons. This is observed as the reactor is started up and the reactor core is still all water (-2)*.

When boiling begins, saturated steam forms and eventually composes approximately 20% of the core. Since the steam is even less dense than the hot water, its moderating ability is further reduced. The corresponding decrease in k_{eff} is observed in Table 9, time step (-1)*. A greater decrease in k_{eff} is caused by Xe^{135} and Sm^{149} buildup (0)*. So, even before the reactor has produced any substantial power, k_{eff} has been decreased by a sizable amount due to the change in moderator density and buildup of fission products.

The reactor is then allowed to run 10 time steps of 73 days each which coincides with an average burnup of 1000 MWD/MT at full power. A complete list of $k_{eff's}$ and computer times for each

* Numbers in parenthesis refer to conditions of the reactor in Table 9 and Figure 7.

reactor condition are presented in Table 9 and k_{eff} over lifetime is plotted in Figure 7.

There are two sets of input values for the FLARE code. The first set was obtained from Commonwealth Edison Company[40] and contains a burnable poison (Tables 10 and 11). This explains the lower k_{eff} at the beginning of the reactor life. After a short period of time, the poison's neutron absorbing ability is decreased to an insignificant amount. In the remainder of the fuel cycle, k_{eff} is primarily a function of fissile isotope mass and fission product buildup.

The second set of input values (Table 12) was obtained from work performed by Combustion Engineering[20]. This set does not include burnable poison since their studies involved only heavy isotope content over core life. The largest disagreement between FEVER and EXTERMINATOR results and FLARE results is in the criticality after time step (6)*. This seems to be due to their individual treatment of fissile plutonium buildup.

Comparison of these theoretical k_{eff} values with actual operating values would be the ideal correlation. The latter values, however, were not available. Therefore, a relative comparison among the codes was made using FEVER values as the basis. FLARE would have been used if the input had been calculated directly. The percentage differences for EXTERMINATOR and FLARE for k_{eff} VS burnup are recorded in Table 13.

TABLE 9 k_{eff} OVER REACTOR LIFE

Computer TIME given in seconds

REACTOR CONDITION	FEVER	TIME	EXTERMINATOR	TIME	FLARE(1)*	TIME	FLARE(2)*	TIME
-3 Cold	1.1884	73	1.1874	933	1.0642	480		
-2 Hot (all water)	1.1469	68	1.1452	472				
-1 No F.P.	1.1349	↑	1.1329	312	1.0468	↑	1.1307	↑
STEPS 0	1.0963		1.0943	306		3409		
1	1.0937		1.0919	347	1.0987		1.1234	
2	1.0856		1.0838	323	1.0842		1.1132	
3	1.0744		1.0727	339	1.0695		1.1013	
4	1.0631		1.0613	291	1.0547	1181	1.0883	1375
5	1.0529		1.0511	259	1.0397		1.0746	
6	1.0447	3063 avg. 255/step	1.0429	319	1.0244		1.0606	
7	1.0385		1.0365	263	1.0089		1.0463	
8	1.0340		1.0320	194	0.9933		1.0318	
9	1.0312		1.0290	333	0.9775		1.0171	
10	1.0295	↓	1.0272	340	0.9617	↓	1.0025	↓

* FLARE (1) signifies constants from Commonwealth Edison were input
FLARE (2) signifies constants from Combustion Engineering were input

FIGURE 7 CRITICALITY VS LIFETIME

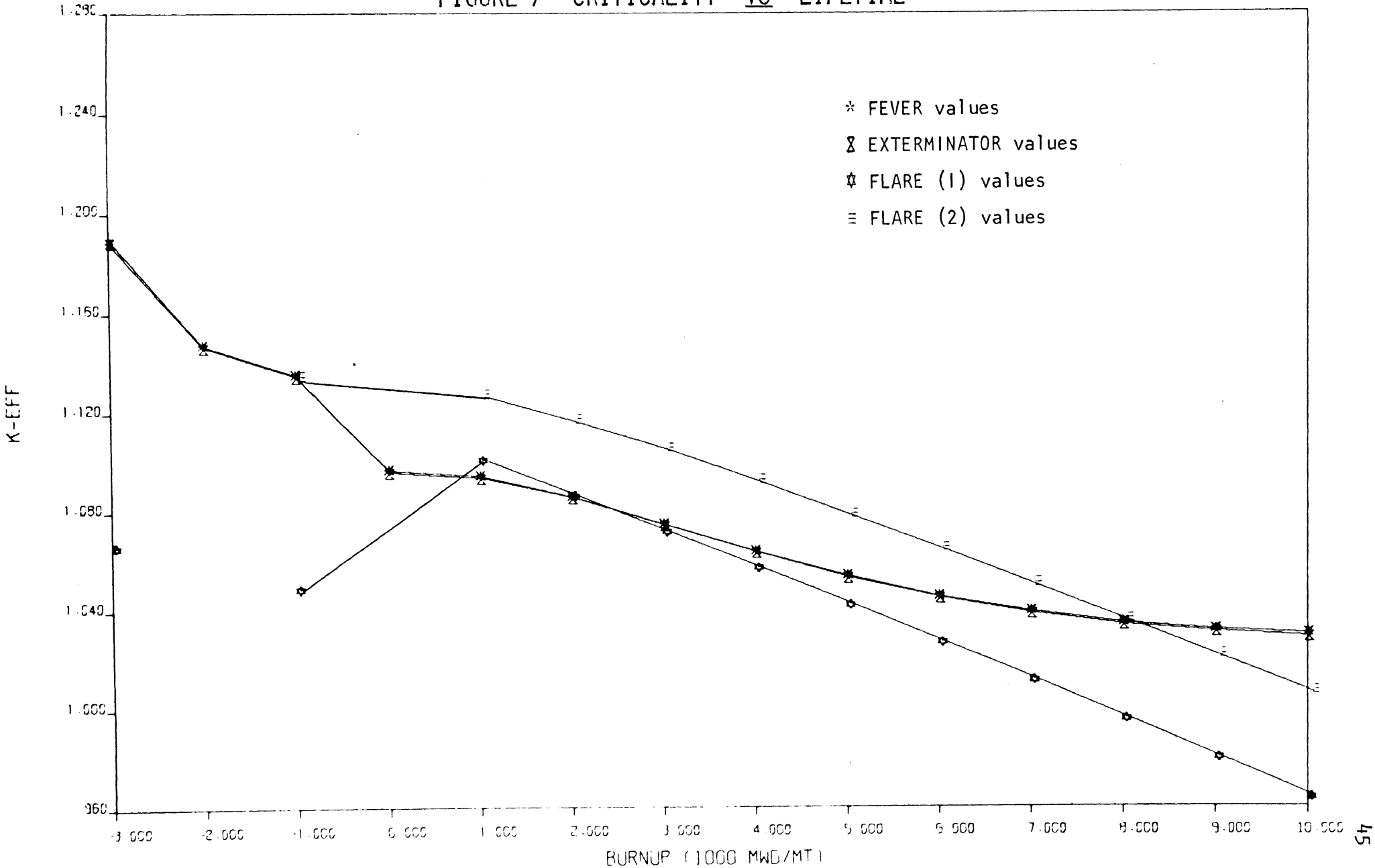


TABLE 10 FLARE CONSTANTS - COLD CASE

Source: COMMONWEALTH EDISON[40]

FLARE INPUT CONSTANTS

$B_1 = 39.80$

$B_2 = 0$

$B_3 = 0$

$B_4 = 1.132$

$B_5 = 0.946$

$B_6 = 80.273$

$B_7 = 0$

$B_8 = 0$

$B_9 = -0.6913$

$B_{10} = 0$

$B_{11} = 0$

$B_{12} = 0.01801$

$B_{13} = 0.0598$

$B_{14} = 0.0410$

$B_{15} = 0.33$

$B_{16} = 0$

$B_{17} = 0$

$B_{18} = 0$

$B_{19} = 0$

$B_{20} = 0.010669$

$B_{21} = 0.275581$

$B_{22} = 1.0$

$B_{23} = 1.0$

$B_{24} = -0.093$

$B_{25} = -0.332$

$B_{26} = 0$

$C_1 = 0$

$C_2 = 0$

$C_3 = 0$

$C_4 = 0$

$C_5 = 0$

$C_6 = 1.0$

$C_7 = 1.0 \times 10^6$

$C_8 = 1.0$

$C_9 = 1.0$

TABLE 11 FLARE CONSTANTS - OPERATING CASE

Source: COMMONWEALTH EDISON[40]

FLARE INPUT CONSTANTS

B_1	154.156	B_{19}	0
B_2	-139.582	B_{20}	0.010669
B_3	46.94	B_{21}	0.275
B_4	1.1121	B_{22}	1.0
B_5	0.7263	B_{23}	1.0
B_6	0.273	B_{24}	-0.093
B_7	0.0886	B_{25}	-0.332
B_8	0.2350	B_{26}	0
B_9	-0.6913		
B_{10}	-0.0382	C_1	0.55
B_{11}	0	C_2	0.8
B_{12}	0.01801	C_3	0
B_{13}	0.0598	C_4	0.525
B_{14}	0.410	C_5	0
B_{15}	0.33	C_6	0.0366
B_{16}	0.0263	C_7	649.4
B_{17}	0.0068	C_8	46.3
B_{18}	1.501	C_9	2.24

TABLE 12 FLARE CONSTANTS - OPERATING CASE

Source: COMBUSTION ENGINEERING[20]

FLARE INPUT CONSTANTS

B_1	= 159.338	B_{19}	= 0
B_2	=-164.059	B_{20}	= 0.010669
B_3	= 61.120	B_{21}	= 0.275581
B_4	= 1.037334	B_{22}	= 1.0
B_5	= 0.79295	B_{23}	= 1.0
B_6	= 0.548565	B_{24}	= -0.093
B_7	= 0.21651	B_{25}	= -0.332
B_8	= 0.436396	B_{26}	= 0
B_9	= 0.852199		
B_{10}	= -0.089772	C_1	= 0.461
B_{11}	= -0.156157	C_2	= 1.21
B_{12}	= -0.281691	C_3	= 0
B_{13}	= 0.023488	C_4	= 0.461
B_{14}	= 2.697116	C_5	= 0
B_{15}	= 0	C_6	= 0.03247
B_{16}	= 0	C_7	=646.6
B_{17}	= 0	C_8	= 0.74049
B_{18}	= 0	C_9	= 0.03666

TABLE 13 k_{eff} CODE DIFFERENCES OVER REACTOR LIFE

FEVER used as basis

REACTOR CONDITION	FEVER k_{eff}	PERCENT DIFFERENCES		
		EXTERMINATOR	FLARE(1)*	FLARE(2)*
-3	1.1884	-.08	-10.45	
-2	1.1469	-.15		
-1	1.1349	-.18	- 7.76	-0.37
0	1.0963	-.18		
1	1.0937	-.16	+ 0.45	+2.72
2	1.0856	-.17	- 0.13	+2.54
3	1.0744	-.33	- 0.46	+2.50
4	1.0631	-.17	- 0.79	+2.37
5	1.0529	-.17	-1.25	+2.06
6	1.0447	-.17	- 1.94	+1.52
7	1.0385	-.19	- 2.85	+ .75
8	1.0340	-.19	- 3.94	- .21
9	1.0312	-.21	- 5.21	-1.37
10	1.0295	-.17	- 6.59	-2.62

* FLARE (1) signifies input constants from Commonwealth Edison were input
FLARE (2) signifies constants from Combustion Engineering were input

C. THERMAL NEUTRON FLUX CORRELATION

FEVER and EXTERMINATOR codes calculate the four group relative neutron fluxes at each mesh point. FLARE, on the other hand, gives the rate of production of fission energy neutrons at each node. Therefore, the thermal fluxes of FEVER and EXTERMINATOR were compared along with the source strengths of FLARE. FEVER being a 1-dimensional code gives the relative fluxes on a radial line from the center of the core to the outer surface of the pressure vessel. Since EXTERMINATOR is a 2-dimensional code and R-Z geometry was employed, its flux pictures were a function of height and radius. In order to compare EXTERMINATOR'S thermal fluxes to those of FEVER, the EXTERMINATOR values were collapsed in the axial direction by averaging at each radial point. This produced 1-dimensional fluxes as in FEVER.

FLARE was included in the correlation by collapsing the sources in the Z-direction to give the average source for each fuel element. By source is meant the rate of production of fission energy neutrons.

$$S_{ij} = k_{\infty ij} A_{ij}$$ where A_{ij} is the absorption rate for the fuel assembly in row i, column j

It is difficult to obtain the source as a function of radius for this case due to the rectangular geometry (Figure 5) of FLARE. However, since the fuel element arrangement is nearly cylindrical, the sources for each fuel assembly along the first row of the quarter core were used for the comparison. Both Combustion

Engineering and Commonwealth Edison values were used for FLARE.

The beginning of life case with no fission products was used for all codes. Relative neutron fluxes and sources are normalized to the first mesh point (5.107 cm) of EXTERMINATOR. FLARE sources are normalized to the centermost fuel element.

Table 14 gives the normalized 1-dimensional thermal fluxes and source values at their respective distances from the center of the core. These values are also plotted in Figure 8. The large differences between FLARE values and FEVER and EXTERMINATOR values stems from the possible difference in fission energy limits and the manner in which leakage is accounted for. FLARE compensates for leakage through the use of albedos while FEVER and EXTERMINATOR work with the reflecting water region explicitly.

D. CONTROL ROD INSERTION COMPARISON

Criticality as a function of control rod insertion for the three codes was also compared. The beginning of life case without fission product buildup was used throughout this particular study and only the amount of control rod insertion was varied. To simulate control rod insertion, FEVER and EXTERMINATOR allow the homogeneous number density of the poison material to change. This is comparable to adding a control poison solution to a particular region. The central region (#1) is the only one which allows a natural Boron (control rod material) number density to

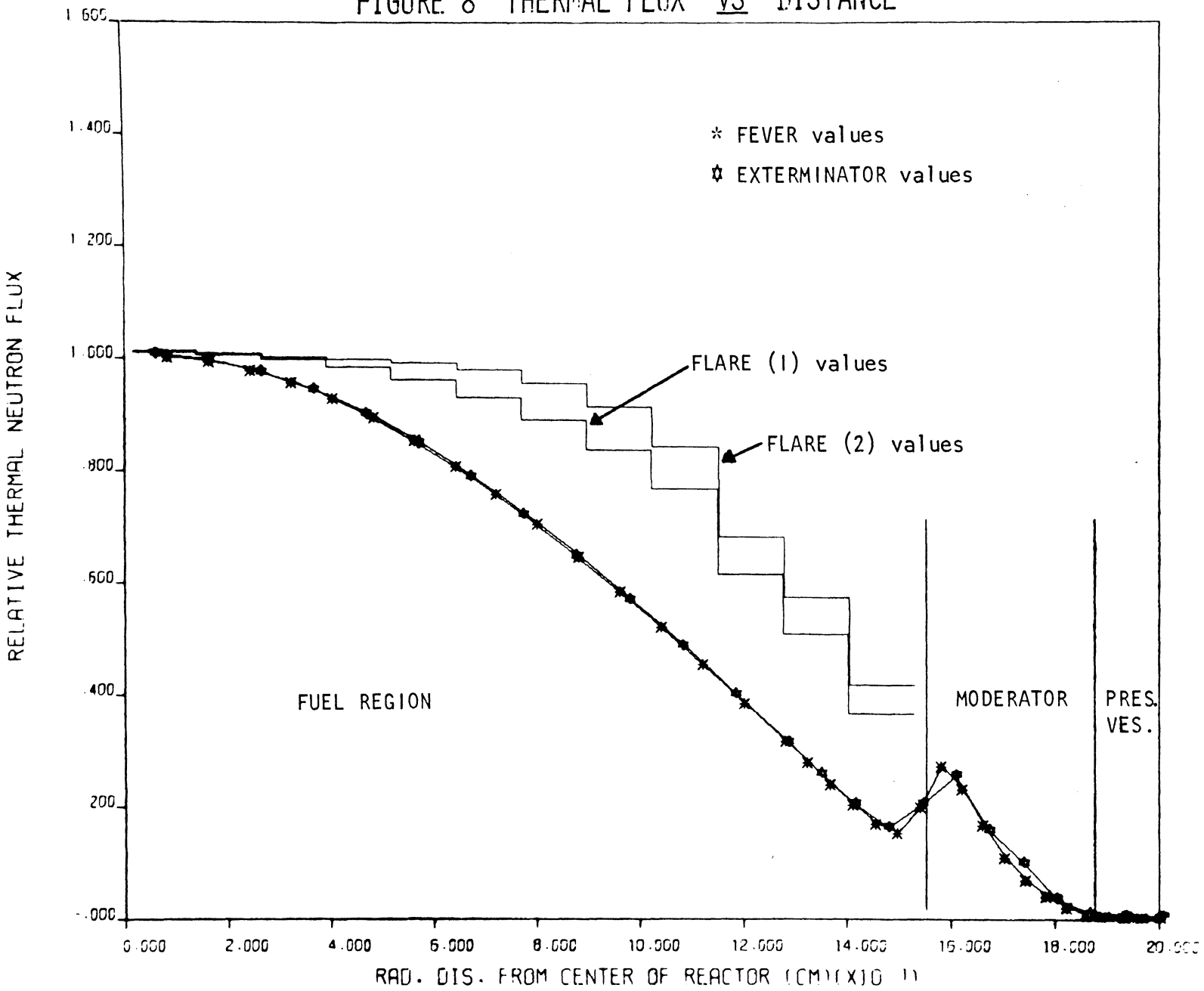
TABLE 14 THERMAL FLUXES, COLLAPSED THERMAL FLUXES, AND SOURCE STRENGTHS (NORMALIZED) VS RADIAL DISTANCE FROM CENTER OF REACTOR CORE

FEVER			EXTERMINATOR			FLARE(1)*		
Pt.	Dis.	Flux	Pt.	Dis.	Flux	Pt.	Dis.	Flux
1	7.981+0	.9976	1	-5.107	1.000	1	12.65	1.0000
2	1.5961+1	.9889	2	5.107	1.0000	2	25.30	0.9928
3	2.3942+1	.9734	3	15.322	0.9889	3	37.95	0.9847
4	3.1922+1	.9515	4	25.537	0.9669	4	50.60	0.9718
5	3.9903+1	.9233	5	35.752	0.9345	5	63.25	0.9485
6	4.7883+1	.8892	6	45.967	0.8922	6	75.90	0.9175
7	5.5864+1	.8494	7	56.182	0.8404	7	88.55	0.8773
8	6.3844+1	.8044	8	66.397	0.7806	8	101.20	0.8255
9	7.1825+1	.7544	9	76.612	0.7132	9	113.85	0.7549
10	7.9805+1	.7000	10	86.827	0.6394	10	126.50	0.6044
11	8.7786+1	.6416	11	97.042	0.5605	11	139.15	0.4963
12	9.5766+1	.5798	12	107.257	0.4779	12	151.80	0.3526
13	1.03746+2	.5153	13	117.472	0.3926			
14	1.11727+2	.4485	14	127.687	0.3063			
15	1.19707+2	.3802	15	134.205	0.2514			
16	1.27688+2	.3111	16	140.722	0.1981			
17	1.32033+2	.2734	17	147.239	0.1564			
18	1.36378+2	.2359	18	153.757	0.1976			
19	1.40722+2	.1992	19	160.191	0.2483			
20	1.45067+2	.1662	20	166.625	0.1514			
21	1.49412+2	.1489	21	173.059	0.0927			
22	1.53757+2	.1962	22	179.493	0.0297			
23	1.57778+2	.2681	23	185.928	0.0029			
24	1.61800+2	.2279	24	192.913	0.0001			
25	1.65821+2	.1628	25	199.898	0.0000			
26	1.69842+2	.1058						
27	1.73864+2	.06454						
28	1.77885+2	.03684						
29	1.81907+2	.01817						
30	1.85928+2	.003777						
31	1.89440+2	.0005853						
32	1.92913+2	.00009263						
33	1.96405+2	.00001535						
34	1.99898+2	.00000237						

FLARE(2)*		
Pt.	Dis.	Flux
1	12.65	1.0000
2	25.30	0.9937
3	37.95	0.9882
4	50.60	0.9831
5	63.25	0.9766
6	75.90	0.9648
7	88.55	0.9420
8	101.20	0.8988
9	113.85	0.8304
10	126.50	0.6676
11	139.15	0.5589
12	151.80	0.4041

* FLARE (1) and FLARE (2) signify constants from Commonwealth Edison and Combustion Engineering respectively.

FIGURE 8 THERMAL FLUX VS DISTANCE



change. This is the primary reason for separating the core into two separate regions in Section III-A.

FLARE, on the other hand, employs individual movement of control rods thus requiring a control rod pattern for each run. Patterns with all rods inserted the same amount (flat pattern) were used in this comparison. Other FLARE control rod schemes are briefly investigated in the next section.

An accurate number density for natural boron in the control rods could not be obtained due to lack of detailed specifications. Therefore, a pseudo number density was used in FEVER and EXTERMINATOR and normalized to 30% rod insertion of FLARE.

Table 15 and Figure 9 gives the k_{eff}'s VS control rod insertion given by the three codes. Their respective computer times are also presented. The values for all three codes are in close agreement up to 50% insertion. The FLARE model affects reactivity more than the FEVER and EXTERMINATOR models for the following reasons: (1) The individual rods in FLARE are being inserted in a high flux region, therefore, higher rod worth and (2) the probability of neutron capture does not increase linearly with homogenized poison number density.

E. POWER DISTRIBUTION STUDY

As previously mentioned, FLARE has the capability of moving control rods independently but does not automatically adjust them for criticality. This must be done by the user. With this

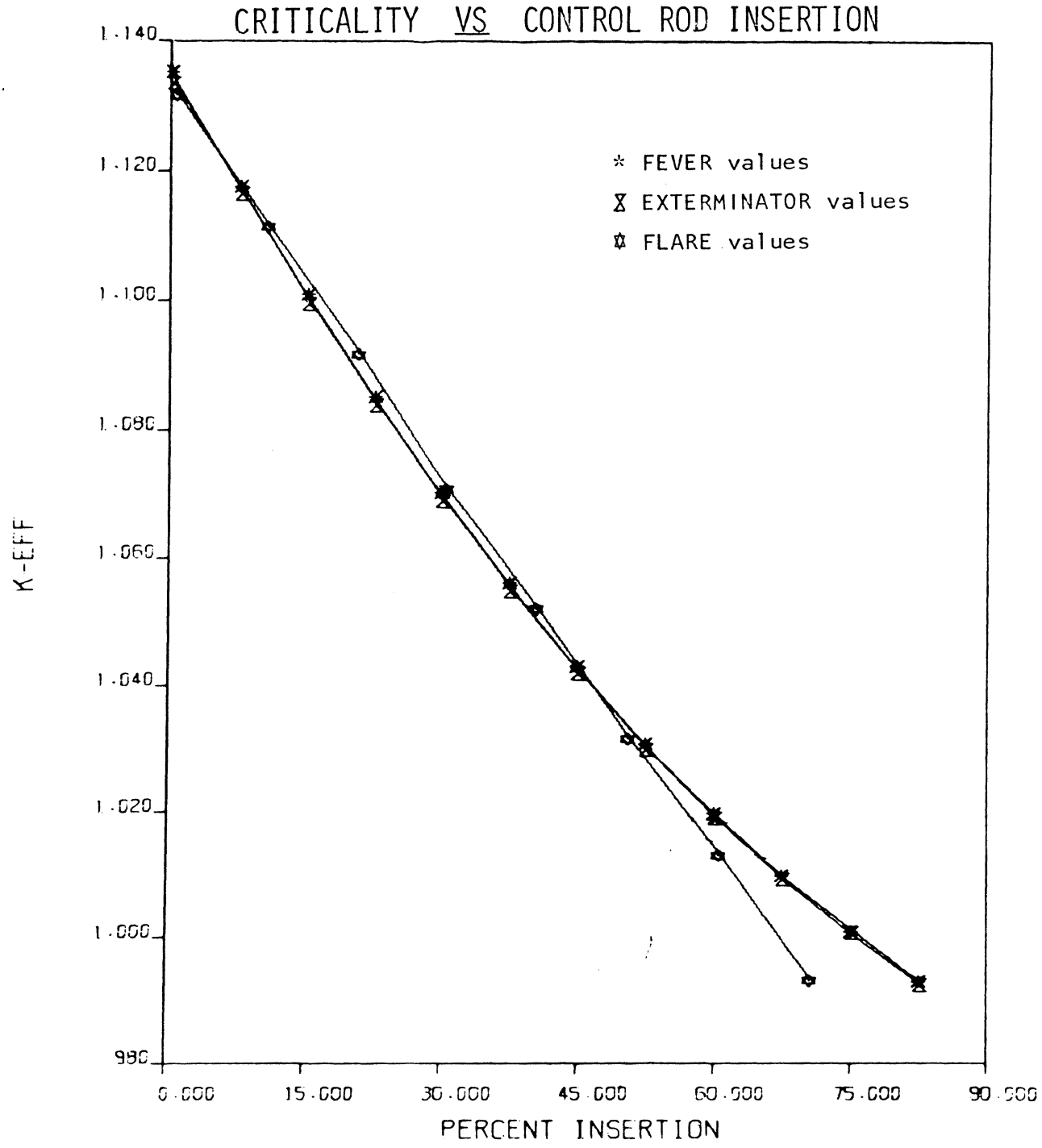
TABLE 15 k_{eff} VS CONTROL ROD INSERTION AT OPERATING BEGINNING OF LIFE

NORMALIZED PERCENT INSERTION	FEVER k_{eff}	CPU TIME(sec)	EXTERMINATOR k_{eff}	CPU TIME(sec)	FLARE(2)* PRECENT INSERTION	FLARE k_{eff}	CPU TIME(sec)
0.0	1.1349	69	1.1329	312	0	1.1307	225
7.5	1.1173	65	1.1166	263	10	1.1104	320**
15.0	1.1006	68	1.0988	↑	20	1.0905	260
22.5	1.0846	65	1.0829	462 ↓	30	1.0694	320**
30.0	1.0695	70	1.0679	↑	40	1.0506	844
37.5	1.0553	64	1.0538		50	1.0301	320**
45.0	1.0421	56	1.0408		60	1.0117	↑
52.5	1.0299	59	1.0288		70	0.9920	873 ↓
60.0	1.0190	54	1.0180	2945			
67.5	1.0091	55	1.0084				
75.0	1.0003	51	0.9999				
82.5	0.9925	45	0.9917	↓			

** combined time of 960

* FLARE (2) signifies constants from Combustion Engineering were input

FIGURE 9


CRITICALITY VS CONTROL ROD INSERTION
1.140
1.120
1.100
1.080
1.060
1.040
1.020
1.000
980
K-EFF
* FEVER values
X EXTERMINATOR values
FLARE values
0.000
15.000
30.000
45.000
60.000
75.000
90.000
PERCENT INSERTION

additional tool at hand, some rod patterns were briefly observed.

The beginning of life case was again singled out for observation. The total amount of control rod insertion was kept constant at 40% while the patterns were altered. This does not necessarily mean that k_{eff} will remain unchanged since the worth of the control rods is a function of individual rod position and amount of insertion.

Three rod patterns were used. They include the flat insertion of the previous section and two attempts at equalizing the average power per assembly. By power flattening is meant to obtain fairly equal power production from all fuel assemblies. This is desirable since it will give a more efficient power production for a loading of uniform enrichment. However, if batches are loaded containing different enrichments, it would be advantageous to have higher thermal flux in assemblies of lower enrichment to achieve maximum burnup. In this case in particular the peak-to-average flux values must be observed in detail to make sure the maximum flux limit is not violated.

In this study it was difficult to control the power in region two since it contains no control rods. To increase the source in this region, the neutron flux would have to be pushed from the inner region by inserting control rods.

The flat control rod pattern in which each rod is inserted

40% of its length can be observed in Figure 10. The respective source strengths by channel are displayed in Figure 11. The source is actually computed at each node producing a three-dimensional picture of power production especially useful in power peaking studies. Axial flattening could also be achieved through the use of partial worth control rods, i.e., rods containing neutron absorbing material in only part of their length with the remainder being a dummy structural material. Partial worth rod constants were not available for this study, therefore, only radial flattening was observed.

The source strengths obtained by the flat insertion pattern were not nearly of equal magnitude. The peak occurs at the reactor center and decreases consistently to the outer fuel assemblies. With these results at hand, a cone shaped control rod pattern (Figure 12) was attempted which resulted in a source depression (Figure 13) in the center assemblies as well as the outer ones. A third pattern (Figure 14) was then used which was also cone shaped but with less of a slope. This produced much more favorable results (Figure 15). With more trials of this type, radial power flattening could be attained with a predetermined amount of accuracy.

FIGURE 10 FLAT CONTROL ROD PATTERN

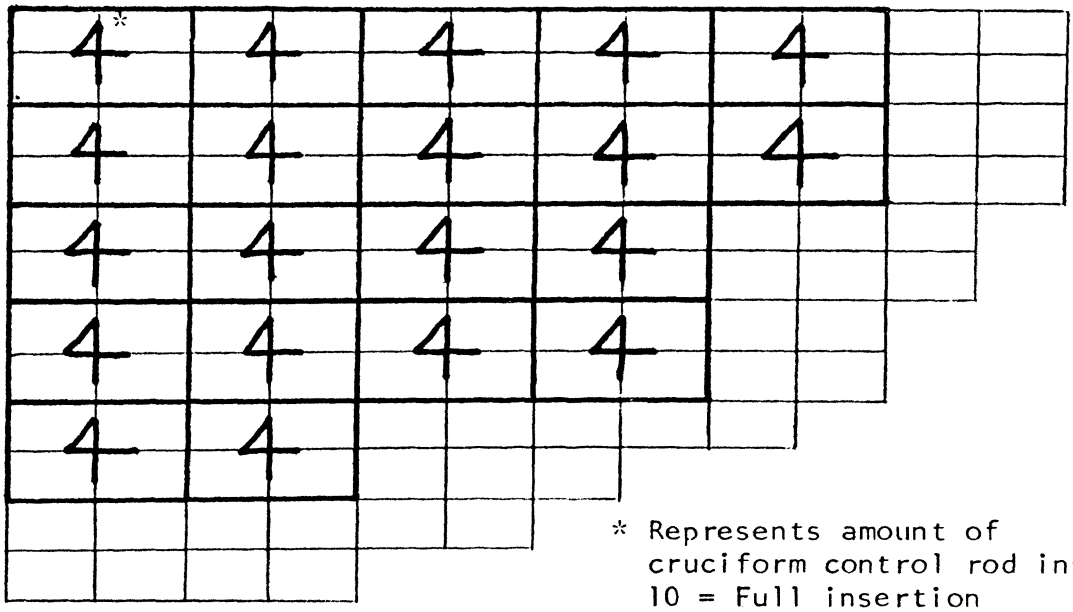


FIGURE 11 SOURCE STRENGTHS PER ELEMENT WITH FIGURE 10 ROD PATTERN

1.58	1.57	1.54	1.50	1.44	1.37	1.28	1.18	1.06	.84	.69	.48
1.57	1.55	1.52	1.48	1.43	1.36	1.27	1.17	1.04	.73	.68	.46
1.54	1.52	1.50	1.45	1.40	1.33	1.24	1.14	1.01	.79	.64	.42
1.50	1.48	1.46	1.41	1.36	1.28	1.20	1.10	.88	.80	.58	.32
1.44	1.43	1.40	1.36	1.30	1.23	1.14	1.03	.83	.68	.47	
1.37	1.36	1.33	1.29	1.23	1.16	1.06	.94	.75	.59	.35	
1.29	1.27	1.24	1.20	1.14	1.06	.96	.77	.65	.46		
1.19	1.18	1.15	1.10	1.03	.94	.77	.65	.51	.31		
1.08	1.06	1.02	.88	.83	.74	.63	.47	.32			
.85	.91	.87	.75	.72	.57	.40					
.70	.69	.65	.58	.47	.34						
.49	.47	.42	.33								

FIGURE 12 STEEP SLOPE CONE SHAPED CONTROL ROD PATTERN

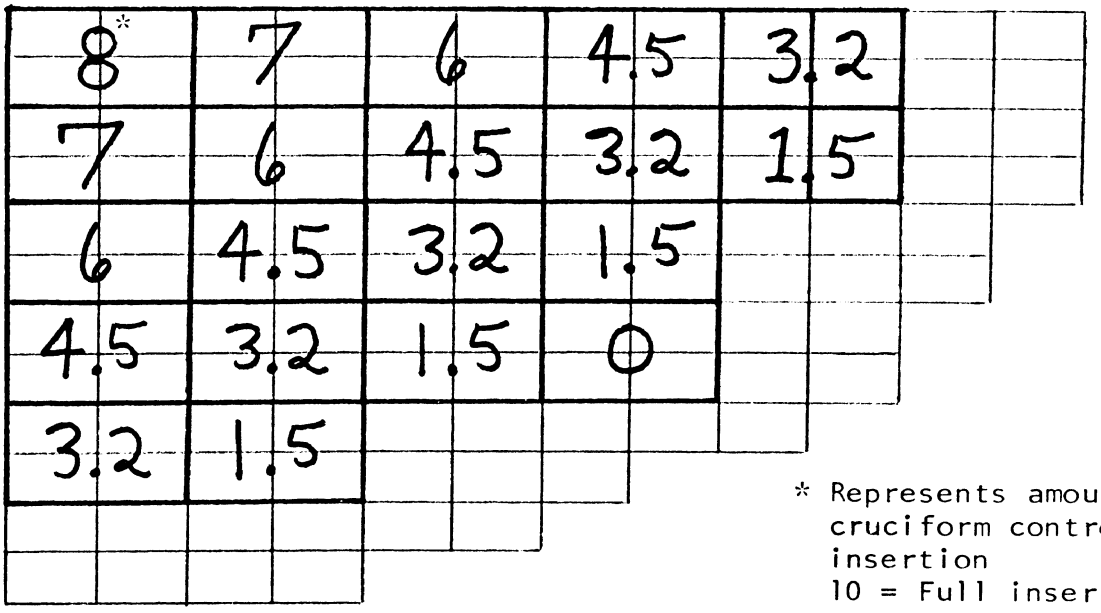


FIGURE 13 SOURCE STRENGTHS PER ELEMENT WITH FIGURE 12 ROD PATTERN

.40	.45	.58	.72	.92	1.06	1.18	1.23	1.20	.98	.83	.62
.45	.49	.62	.78	.96	1.11	1.22	1.27	1.21	.92	.82	.61
.57	.62	.77	.92	1.08	1.20	1.32	1.32	1.25	.99	.82	.57
.72	.78	.92	1.04	1.18	1.32	1.38	1.37	1.15	1.07	.78	.46
.92	.96	1.08	1.18	1.33	1.41	1.46	1.40	1.12	.93	.70	
1.06	1.11	1.20	1.32	1.40	1.46	1.46	1.34	1.05	.84	.55	
1.19	1.23	1.32	1.38	1.45	1.45	1.37	1.10	.93	.72		
1.27	1.29	1.34	1.37	1.38	1.32	1.09	.95	.78	.52		
1.24	1.26	1.28	1.15	1.11	1.02	.90	.75	.53			
1.05	1.12	1.13	.99	1.00	.82	.64					
.88	.88	.85	.79	.70	.53						
.66	.65	.60	.48								

FIGURE 14 MODERATE SLOPE CONE SHAPED CONTROL ROD PATTERN

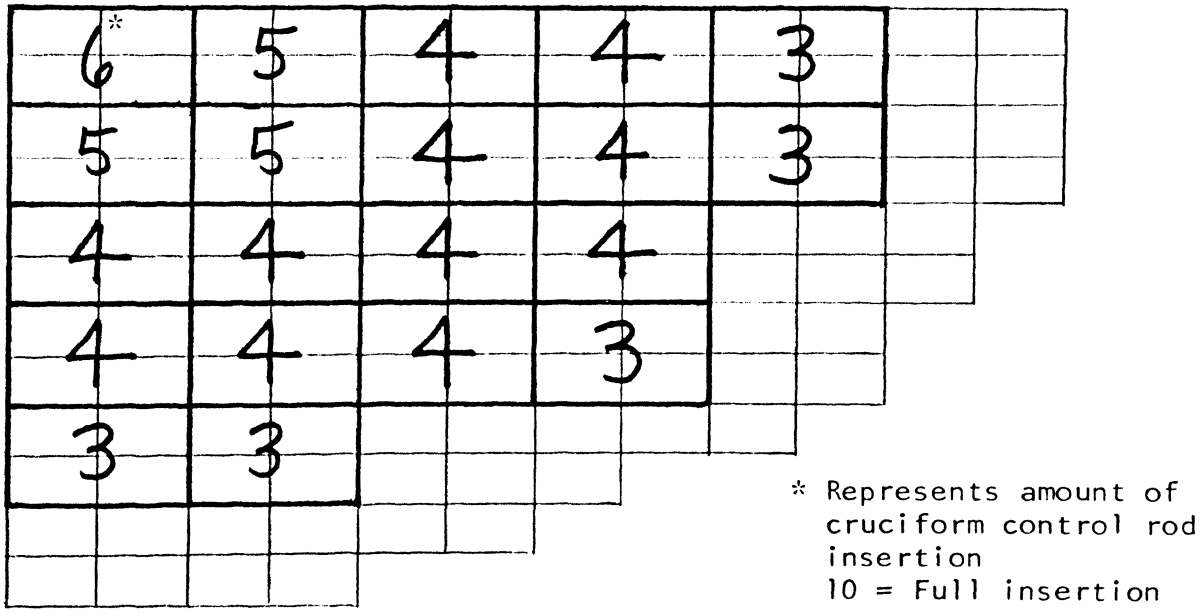


FIGURE 15 SOURCE STRENGTHS PER ELEMENT WITH FIGURE 14 ROD PATTERN

1.34	1.35	1.37	1.37	1.39	1.34	1.27	1.18	1.07	.88	.75	.54
1.35	1.36	1.37	1.37	1.38	1.34	1.26	1.16	1.06	.82	.73	.52
1.37	1.37	1.37	1.36	1.37	1.32	1.24	1.14	1.03	.84	.70	.46
1.37	1.37	1.36	1.35	1.34	1.28	1.20	1.10	.92	.84	.64	.36
1.40	1.39	1.38	1.35	1.30	1.23	1.15	1.05	.86	.73	.52	
1.35	1.34	1.32	1.29	1.24	1.17	1.08	.97	.80	.64	.38	
1.28	1.27	1.25	1.21	1.15	1.09	.99	.84	.71	.50		
1.19	1.18	1.15	1.11	1.05	.97	.84	.72	.56	.34		
1.09	1.07	1.04	.93	.87	.80	.69	.51	.35			
.89	.95	.91	.81	.77	.63	.43					
.76	.94	.71	.65	.52	.38						
.55	.53	.48	.37								

V. CONCLUSIONS

For minute flux studies where computer time is not heavily restricted, a two-dimensional code utilizing as many mesh points as possible yields good results. Two-dimensional R-Z geometry is best for a cylindrical core that has angular uniformity. Theoretically, working in three dimensions would be better but few codes permit much detail due to the large computer memory requirement.

On the other hand, macroscopic values such as k_{eff} can be found with reasonable accuracy using a one-dimensional diffusion code. For an identical mesh spacing, the two- and three-dimensional codes yield slightly more reliable values mainly because of their more accurate consideration of neighboring effects. The accuracy of the macroscopic values is essentially unchanged by looser convergence criteria on microscopic parameters. This is a big factor when considering computer time.

In reactor lifetime k_{eff} studies, the values from one-, two-, and three-dimensional codes are similar but reliability after 6000 or 7000 MWD/T is doubtful. The FLARE output providing burnup as a function of fuel assembly increment (node) is extremely helpful, especially for use in future fuel cycles where shuffling would be required.

FLARE values appear to be in poor agreement for this study. The primary cause is presumably due to obtaining the input values from independent sources rather than calculating them directly from basic

data. Direct calculation would be quite involved and the facilities for doing so were not available.

VI. RECOMMENDATIONS

A similar criticality over reactor lifetime study should be made employing time spectrum-dependent cross sections. This would basically mean inserting the current mass balances from the diffusion codes into the cross section codes. This would be performed each time step thus obtaining a new set of flux-weighted cross sections for use over the following time step. Automating the flow of data to perform the operations would be another improvement saving much labor and possibility of error.

Three different codes of the same nature could be used for a similar study. Using a two-dimensional code rather than a straight diffusion code would also be more correct and easier to work with. A diffusion code such as PDQ-7 which can work in one, two, or three dimensions would be ideally suited for this type analysis.

Total core flux control using partial worth control rods would provide interesting as well as useful results. The objective may include uniform power production with minimum power peaks. The study could be carried through several fuel cycles.

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APPENDIX I

Computer Program to Punch Weighted FEVER Cross Sections

The computer program described below was created to alleviate the tedious task of combining cross sections of the hot water and saturated steam varieties listed in Table A2. Two sets of cross section blocks must be previously obtained in FEVER format, one set representing the full core as hot water and the other set representing the full core as saturated steam. By combining the blocks linearly, a new set of cross sections are punched. This treatment helps account for cross section dependence on any percentage of steam voiding.

The combining equation is

$$\sigma_{NEW} = (\sigma_{HOT\ WATER})(HWFRT) + (\sigma_{SAT.\ STEAM})(STFRT)$$

Input is arranged as follows:

	COLUMN	
CARD #1	1-3	NBLK, Number of pairs of cross section blocks (I3)
	4-6	NOG, Number of energy groups (I3)
	7-26	Volume fractions of core which are hot water (HWFRT) and saturated steam (STFRT) respectively (2F10.5)
	27-46	Hollerith to be punched on header of each block

CARDS #2 - to end of input

Hot water block for nuclide 1
Saturated steam block for nuclide 1
Hot water block for nuclide 2
Etc.

A sample computer input is given in Table A2 and the resulting output in Table A3.

Table A1

FORTRAN SOURCE DECK LISTING OF WEIGHTED FEVER CROSS SECTION PROGRAM

```
C-------- INPUT/OUTPUT UNITS   L1=READ, L2=PUNCH, L3=WRITE
      L1=1
      L2=2
      L3=3
C        NBLK.....NUMBER OF PAIRS OF CROSS SECTION BLOCKS
C        NOG......TOTAL NUMBER OF GROUPS
C        HWFRT.....FRACTION BY WHICH ODD NUMBERED(HOT WATER) BLOCKS ARE WEIGHTED
C        STFRT.....FRACTION BY WHICH EVEN NUMBERED(SAT.STM.) BLOCKS ARE WEIGHTED
C        CR(4,5)...ODD NUMBERED BLOCKS
C        CRH(4,5)..EVEN NUMBERED BLOCKS
C-------- INPUT BLOCKS MUST BE ARRANGED IN PAIRS
C
      DIMENSION CR(4,5),CRH(4,5)
      READ(L1,8) NBLK,NOG,HWFRT,STFRT,ID1,ID2,ID3,ID4
      WRITE(L3,9)NBLK,NOG,HWFRT,STFRT,ID1,ID2,ID3,ID4
      DO 150 IJJ=1,NBLK
      READ(L1,10) A1,A2,A3,XBUG
      WRITE(L3,13) A1,A2,A3,ID1,ID2,ID3,ID4,XBUG
      WRITE(L2,11) A1,A2,A3,ID1,ID2,ID3,ID4,XBUG
      DO 20 IJ=1,NOG
   20 READ(L1,12) (CR(IJ,J),J=1,5)
      READ(L1,10) XBUG
      DO 30 JJ=1,NOG
   30 READ(L1,12) (CRH(JJ,J),J=1,5)
      DO 40 II=1,NOG
      DO 40 IJ=1,5
   40 CR(II,IJ)=CR(II,IJ)*HWFRT+CRH(II,IJ)*STFRT                    WT. CARD
      DO 50 II=1,NOG
      WRITE(L3,14)  (CR(II,J),J=1,5)
   50 WRITE(L2,12) (CR(II,J),J=1,5)
  150 CONTINUE
    8 FORMAT(2I3,2F10.5,4A4)
    9 FORMAT('1   NUMBER OF PAIRS OF CROSS SECTION BLOCKS =',I3/'     NUM
     1BER OF ENERGY GROUPS =',I3/'     HWFRT =',F6.4,'     STFRT =',F6.4/'
     2    MIXTURE INFORMATION PUNCHED ON FIRST CARD OF EACH BLOCK---',4A
     3//' ------------------------  LISTING OF PUNCHED CARDS  --------
     4-----------------'/)
   10 FORMAT(3A4,48X,1PE12.4)
   11 FORMAT(1X,3A4,25X,4A4,6X,1PE12.4)
   13 FORMAT(2X,3A4,25X,4A4,6X,1PE12.4)
   12 FORMAT(1P5E12.5)
   14 FORMAT(1X,1P5E12.5)
      STOP
      END
```

Table A2

SAMPLE INPUT DATA FOR THE PUNCH PROGRAM IN TABLE A1

```
  4  4       0.8        0.2 HOT DP. .8W-.2ST
U235(L)*          AR030967
 3.42546E+00 5.54455E+00 1.48969E+00 9.12063E-02 2.59033E+00
 9.31133E+00 1.71298E+01 5.32898E+00 3.79631E-02 2.43400E+00
 1.00007E+02 7.55965E+01 5.85810E+01 1.41256E-02 2.43400E+00
 714.66+00   358.07+00      .3491E+03  0.0          2.43+00
U235(L)*          AP030967
 3.43021E+00 5.70181E+00 1.51590E+00 9.93167E-02 2.57228E+00
 9.25352E+00 1.70462E+01 5.29398E+00 3.70630E-02 2.43400E+00
 9.61850E+01 7.46022E+01 5.65692E+01 1.21409E-02 2.43400E+00
 669.95+00   336.6+00      .3266E+03  0.0          2.43+00
U236              A 082666
 1.08161E+00 5.67742E+00 5.92151E-01 1.40056E-01 2.75569E+00
 0.          1.48349E+01 1.28501E+00 3.21918E-02 0.
 0.          7.22992E+01 4.03874E+01 9.93522E-03 0.
  0.0        13.832+00      .3860E+01  0.0          0.0
U236              A 082666
 9.58301E-01 5.83235E+00 5.56753E-01 1.52886E-01 2.74523E+00
 0.          1.48015E+01 1.26959E+00 3.14268E-02 0.
 0.          7.24763E+01 3.79823E+01 8.53924E-03 0.
 0.0          13.61+00      .3638E+01  0.0          0.0
U238*             A 120266
 5.72426E-01 5.69341E+00 3.19816E-01 1.71697E-01 2.79603E+00
 0.          1.51465E+01 6.24993E-01 3.70502E-02 0.
 0.          2.49082E+01 2.51929E+00 8.36555E-03 0.
  0.0         9.561+00      .1494E+01  0.0          0.0
U238*             A 120266
 5.00160E-01 5.85002E+00 3.02255E-01 1.85315E-01 2.78621E+00
 0.          1.51011E+01 6.23067E-01 3.61698E-02 0.
 0.          2.25374E+01 2.33359E+00 7.19012E-03 0.
 0.0          9.475+00      .1408E+01  0.0          0.0
NP239    101
 3.04548E+00 6.53551E+00 1.21520E+00 8.98289E-03 2.54000E+00
 3.09140E-01 1.06422E+01 1.85805E-01 2.05383E-02 2.54000E+00
 2.30869E-01 1.22341E+01 1.45486E+00 1.00393E-02 2.54000E+00
  0.0        27.805+00      .1786E+02  0.0          0.0
NP239    101
 2.98913E+00 6.69596E+00 1.19401E+00 1.05665E-02 2.54000E+00
 3.14484E-01 1.06411E+01 1.87570E-01 2.00502E-02 2.54000E+00
 2.12973E-01 1.21681E+01 1.34213E+00 8.62867E-03 2.54000E+00
 0.0          27.805+00      .1786E+02  0.0          0.0
```

```
1.0000E+00U235(   0
           U235(   1
           U235(   2
           U235(   3
           U235THHW
1.0000E+00U235(   0
           U235(   1
           U235(   2
           U235(   3
           U235THST
1.0000E+00U236    0
           U236    1
           U236    2
           U236    3
           U236THHW
1.0000E+00U236    0
           U236    1
           U236    2
           U236    3
           U236THST
1.0000E+00U238*   0
           U238*   1
           U238*   2
           U238*   3
           U238THHW
1.0000E+00U238*   0
           U238*   1
           U238*   2
           U238*   3
           U238THST
1.0000E+00NP239   0
           NP239   1
           NP239   2
           NP239   3
           N239THHW
1.0000E+00NP239   0
           NP239   1
           NP239   2
           NP239   3
           N239THST
```

Table A3

SAMPLE COMPUTER OUTPUT FOR INPUT IN TABLE A2

```
   NUMBER OF PAIRS OF CROSS SECTION BLOCKS =  4
   NUMBER OF ENERGY GROUPS =  4
   HWRT =0.8000     STRT =0.2000
   MIXTURE INFORMATION PUNCHED ON FIRST CARD OF EACH BLOCK---HOT OP. .8W-.2ST

------------------------- LISTING OF PUNCHED CARDS -------------------------

U235(L)*                                          HOT OP. .8W-.2ST          1.0000E 00
3.42641E 00 5.57600E 00 1.49495E 00 9.28283E-02 2.58672E 00
9.29977E 00 1.71131E 01 5.32198E 00 3.77831E-02 2.43400E 00
9.92424E 01 7.53976E 01 5.81786E 01 1.37287E-02 2.43400E 00
7.05719E 02 3.53776E 02 3.43800E 02 0.0         2.43000E 00
U236                                              HOT OP. .8W-.2ST          1.0000E 00
1.05695E 00 5.70840E 00 5.85071E-01 1.42622E-01 2.75360E 00
0.0         1.48282E 01 1.28192E 00 3.20388E-02 0.0
0.0         7.23346E 01 3.99064E 01 9.65602E-03 0.0
0.0         1.37976E 01 3.81560E 00 0.0         0.0
U238*                                             HOT OP. .8W-.2ST          1.0000E 00
5.57973E-01 5.72473E 00 3.16304E-01 1.74420E-01 2.79406E 00
0.0         1.51374E 01 6.24608E-01 3.68741E-02 0.0
0.0         2.44340E 01 2.48215E 00 8.13046E-03 0.0
0.0         9.54380E 00 1.47680E 00 0.0         0.0
NP239    101                                      HOT OP. .8W-.2ST          1.0000E 00
3.03421E 00 6.56760E 00 1.21096E 00 9.29961E-03 2.54000E 00
3.10209E-01 1.06420E 01 1.86158E-01 2.04407E-02 2.54000E 00
2.27291E-01 1.22209E 01 1.43231E 00 9.75717E-03 2.54000E 00
0.0         2.78050E 01 1.78600E 01 0.0         0.0
```

APPENDIX II

Computer Program to Punch EXTERMINATOR-2 Cross Sections from FEVER code

The computer program written to punch EXTERMINATOR-2 cross sections is listed in Table A4. FEVER library cross sections are read in, rearranged, and punched on cards in a manner compatible with EXTERMINATOR-2 formats. The program rounds each value to four significant figures since EXTERMINATOR-2 reads cross sections with an E9.0 format.

Input is arranged as follows:

CARD 1 COL. 1-3 NBLK, Number of cross section blocks from FEVER library (13)

COL. 4-6 NOG, Number of energy groups (13)

CARD 2 - to end of input

Block 1 from FEVER library

Block 2 from FEVER library

Block 3 from FEVER library

Etc.

where a block represents the group dependent cross sections for a particular nuclide.

A sample computer input is given in Table A5 and the resulting output in Table A6.

Table A4

FORTRAN SOURCE DECK LISTING OF EXTERMINATOR PUNCH PROGRAM

```
C
C-------- INPUT/OUTPUT UNITS  L1=READ, L2=PUNCH, L3=WRITE
      L1=1
      L2=2
      L3=3
C        NBLK......NUMBER OF CROSS SECTION BLOCKS FROM FEVER
C        NOG.......TOTAL NUMBER OF GROUPS
C        IJJ.......NUCLIDE IDENTIFICATION NUMBER
C        II........GROUP NUMBER
C        CR(1).....NU*MICROSCOPIC FISSION CROSS SECTION
C        CR(2).....MICROSCOPIC TRANSPORT CROSS SECTION
C        CR(3).....MICROSCOPIC ABSORPTION CROSS SECTION
C        CR(4).....MICROSCOPIC SCATTER OUT CROSS SECTION
C        CR(5).....NU = NEUTRONS/FISSION
C
      DIMENSION CR(5)
      READ(L1,8) NBLK,NOG
      WRITE(L3,9) NBLK,NOG
      DO 150 IJJ=1,NBLK
      READ(L1,10) A1,A2
      DO 155 II=1,NOG
      READ(L1,11) (CR(I),I=1,5),A1,A2
      WRITE(L3,120) IJJ,II,A1,A2
      WRITE(L2,12) IJJ,II,A1,A2
C--------CHECK FOR NU=0.0
      IF(CR(5)) 20,21,20
   21 CR(1)=0.0
      GO TO 25
   20 CR(1)=CR(1)/CR(5)
C--------CR(1) IS NOW MICROSCOPIC FISSION CROSS SECTION
   25 WRITE(L2,13) CR(3),CR(1),CR(2),CR(5)
      WRITE(L3,130) CR(3),CR(1),CR(2),CR(5)
      IF(II-2) 30,31,32
   30 WRITE(L2,14) CR(4)
      WRITE(L3,140) CR(4)
      GO TO 155
   31 WRITE(L2,15) CR(4)
      WRITE(L3,151) CR(4)
      GO TO 155
   32 WRITE(L2,16) CR(4)
      WRITE(L3,160) CR(4)
  155 CONTINUE
  150 CONTINUE
    8 FORMAT(2I3)
    9 FORMAT('1  NUMBER OF CROSS SECTION BLOCKS =',I3,'   NUMBER OF ENER
     XGY GROUPS =',I3/' ------------------------  LISTING OF PUNCHED CA
     XRDS  -------------------------'/)
   10 FORMAT(1X,2A4)
   11 FORMAT(1P5E12.4,12X,2A4)
```

```
 12 FORMAT(2I3,66X,2A4)
120 FORMAT(1X,2I3,66X,2A4)
 13 FORMAT(2P4E9.2)
130 FORMAT(1X,2P4E9.2)
 14 FORMAT(9X,2PE9.2)
140 FORMAT(1X,9X,2PE9.2)
 15 FORMAT(18X,2PE9.2)
151 FORMAT(1X,18X,2PE9.2)
 16 FORMAT(27X,2PE9.2)
160 FORMAT(1X,27X,2PE9.2)
    STOP
    END
```

Table A5

SAMPLE INPUT DATA FOR THE PUNCH PROGRAM IN TABLE A4

```
 4  4
U235(L)*          ARC30967       COLD                                    1.0000E+00U235(
3.42552E+00 5.50785E+00 1.48348E+00 8.92423E-02 2.59527E+00                       U235(   1
9.38325E+00 1.71770E+01 5.37316E+00 3.81885E-02 2.43400E+00                       U235(   2
1.00512E+02 7.55360E+01 5.87376E+01 1.46259E-02 2.43400E+00                       U235(   3
.104539E+04  .51547E+03  .50550E+03  0.0         2.43+00                        U235THCD
U236              A C82666       COLD                                    1.0000E+00U236
1.11415E+00 5.64098E+00 6.01258E-01 1.36938E-01 2.75879E+00                       U236    1
0.          1.48401E+01 1.28854E+00 3.23832E-02 0.                            U236    2
0.          7.21847E+01 4.09079E+01 1.02871E-02 0.                            U236    3
0.0          .15341E+02   .5369E+01  0.0          0.0                         U236THCD
U238*             A 120266       COLD                                    1.0000E+00U238*
5.91896E-01 5.65659E+00 3.24610E-01 1.68351E-01 2.79898E+00                       U238*   1
0.          1.51401E+01 6.08957E-01 3.72706E-02 0.                            U238*   2
0.          2.56429E+01 2.45167E+00 8.66181E-03 0.                            U238*   3
0.0          .10145E+02   .2078E+01  0.0          0.0                         U238THCD
NP239    101                       COLD                                    1.0000E+00NP239
3.05991E+00 6.49798E+00 1.22063E+00 8.62707E-03 2.54000E+00                       NP239   1
3.08000E-01 1.06422E+01 1.85434E-01 2.06605E-02 2.54000E+00                       NP239   2
2.35301E-01 1.22450E+01 1.48279E+00 1.03948E-02 2.54000E+00                       NP239   3
0.0          .32692E+02   .2272E+02  0.0          0.0                         N239THCD
```

Table A6

SAMPLE COMPUTER OUTPUT FOR INPUT DATA IN TABLE A5

```
  NUMBER OF CROSS SECTION BLOCKS =  4    NUMBER OF ENERGY GROUPS =  4
------------------------- LISTING OF PUNCHED CARDS -------------------------

  1  1
14.83E-0113.20E-0155.08E-0125.95E-01                                  U235(  1
         89.24E-03
  1  2
53.73E-0138.55E-0117.18E 0024.34E-01                                  U235(  2
                   38.19E-03
  1  3
58.74E 0041.29E 0075.54E 0024.34E-01                                  U235(  3
                                14.63E-03
  1  4
50.55E 0143.02E 0151.55E 0124.30E-01                                  U235THCD
                                  0.0
  2  1
60.13E-0240.39E-0256.41E-0127.59E-01                                  U236   1
         13.69E-02
  2  2
12.89E-01 0.0      14.84E 00 0.0                                      U236   2
                   32.38E-03
  2  3
40.91E 00 0.0      72.18E 00 0.0                                      U236   3
                                10.29E-03
  2  4
53.69E-01 0.0      15.34E 00 0.0                                      U236THCD
                                  0.0
  3  1
32.46E-0221.15E-0256.57E-0127.99E-01                                  U238*  1
         16.84E-02
  3  2
60.90E-02 0.0      15.14E 00 0.0                                      U238*  2
                   37.27E-03
  3  3
24.52E-01 0.0      25.64E 00 0.0                                      U238*  3
                                86.62E-04
  3  4
20.78E-01 0.0      10.14E 00 0.0                                      U238THCD
                                  0.0
  4  1
12.21E-0112.05E-0164.98E-0125.40E-01                                  NP239  1
         86.27E-04
  4  2
18.54E-0212.13E-0210.64E 0025.40E-01                                  NP239  2
                   20.66E-03
  4  3
14.83E-0192.64E-0312.24E 0025.40E-01                                  NP239  3
                                10.39E-03
  4  4
22.72E 00 0.0      32.69E 00 0.0                                      N239THCD
                                  0.0
```

VITA

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He has been enrolled in the Graduate School of the University of Missouri at Rolla since September, 1967, and has held an Atomic Energy Commission Traineeship for the period September, 1967, to March, 1969.