One dimensional depletion analysis of a research reactor

by

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I. INTRODUCTION

Fuel depletion analysis is concerned with predicting the long-term changes in reactor fuel composition caused by nuclei transmutation during reactor operation. Such changes have an important bearing on the operating life of a reactor, as well as on its stability and control. One must first ensure that the shift in the core power distribution that accompanies fuel burnup does not result in the exceeding of core thermal limitations. Sufficient excess reactivity must be provided in the fresh core loading to achieve the desired fuel exposure consistent with safety limitations. A detailed analysis of core composition is necessary in order to optimize fuel exposure to achieve minimum power costs as well as to determine the value of discharged fuel. Since fuel costs over the operating lifetime of the reactor can exceed those of the capital cost of the plant itself, the incentive for accurate analysis of fuel depletion is quite high.

A variety of nuclear processes must be monitored during a depletion study. These include the consumption of fissile nuclides (fuel burnup). However one must also account for the conversion of fertile isotopes into fissile isotopes and the production of numerous fission products. Finally, one must monitor the reactivity balance to ensure core criticality. This is usually done by determining the change in reactivity

over a period of core operation and then adjusting control to compensate for this reactivity change.

Since depletion studies are so complex, models must be developed to simplify calculations. In this study, a one-dimensional model suggested by Dr. S. H. Levine[1] of the Pennsylvania State University is used to calculate the core burnup of the Ames Laboratory Research Reactor (ALRR).

The method consists of using the LEOPARD code as modified for plate type fuel elements to generate two group macroscopic cross sections as functions of burnup in megawatt days per metric tonne uranium (MWD/MTU or J/kg) (1). These macroscopic parameters are then used as input data to the one-dimensional diffusion theory code FOG to generate the flux profiles at each depletion step. Equations suggested by Levine were developed and then used to calculate the one-dimensional cumulative burnup by dividing the cylindrical ALRR core into annular rings.

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[1]S. H. Levine, Dept. of Nuclr. Engrg., PSU. In-core fuel management. Unpublished mimeographed paper, 1976.

II. LITERATURE REVIEW

Although the subjects of depletion analysis and depletion calculational codes for power reactors are extensively treated in the literature, little computational work appears to have been published on depletion studies of research reactors.

Probably the best references in this area are reported by Levine in which a discussion of core management models (2) and computer programs for fuel management (3) for the TRIGA-type (4) research reactor are presented. In the core management model, the burnup of each fuel element was followed individually. Also, a subroutine generated two group macroscopic cross sections using polynomial expansions and thus eliminated the need for LEOPARD calculations. Furthermore, the mathematical scheme developed for the TRIGA core management model is reported to have great potential for application to power reactors.

Information is also available concerning the fuel cycles in heavy water research reactors of the type of the ALRR (5). They depend on the power level at which the reactor is operated, the loading pattern, and the fuel loading per element. In one scheme, used at the MIT heavy water research reactor, only a few of the fuel elements were replaced at a time. In another scheme, operations were started with the required fuel mass, and elements were added to maintain

reactivity. Typical burnups attained in heavy water research reactors are reported to be between 30 and 40 per cent of the U-235.

The ALRR (6) can be operated continuously at 5.0 MW for approximately 170 days without shutdown. The average U-235 burnup for this type of operation will be 30 grams of U-235 per fuel element or 17.7 atom per cent burnup. A greater U-235 burnup can be achieved by either programming the fuel element rotation or by using some of the reactivity allowed for experiments to compensate for the U-235 burnup and low cross section fission products. A report by McCorkle and Norman (7) gives the burnup of an average ALRR fuel element to be 20.00 atom per cent.

Typical values of the radial and axial neutron flux distribution for a heavy water reactor are available (5). For a given power, the heavy water reactor has a higher thermal neutron flux than any other type of reactor. The good reflecting properties of the heavy water and graphite produce a rather flat neutron flux distribution across the core.

III. THEORY

The power distribution across a nuclear reactor core is never flat, so that fuel assemblies produce power, P_j, at different magnitudes. The power distribution as represented by the relative power (RP) of each fuel assembly, RP_j, at the beginning of the core life establishes the essential data needed to determine the isotopic change in each fuel assembly as the core depletes. Those fuel assemblies having $RP_j > 1$ deplete faster than the fuel assemblies producing the average power,

$$P_j = \bar{P} \quad \text{or} \quad RP_j = 1.0$$

while those with $RP_j < 1$ deplete more slowly. The depletion of each fuel assembly is calculated as a function of its burnup, BU_j.

Let

$\overline{BU}(d)$ = burnup of the average fuel assembly after operating d days at full power

$BU(d)_j$ = corresponding burnup of the jth fuel assembly

The burnup $BU(d)_j$ is defined as

$$BU(d)_j = P_j d$$

or

$$BU(d)_j = (RP_j) \cdot \bar{P} \cdot d.$$

The units of burnup are usually megawatt days per metric tonne uranium (MWD/MTU or J/KG). Burnup represents the total amount of heat which is released per unit mass of fuel fed. The core depletion is calculated over separate steps $p \cdot d_i$ where d_i is equal to the operating time d divided by the mass of fuel fed. In general, d_i is taken small until equilibrium Xenon is reached after which d_i is increased. Table 1 provides an example of a depletion schedule for the inner ring of fuel elements of the ALRR. During each time step, it is assumed in the calculation that the power distribution remains constant.

Let

$BU(d_i)_j$ = the burnup of the j^{th} fuel assembly during time step d_i

$\overline{BU}(d_i)$ = the burnup of the average fuel assembly during time step d_i.

Then the cumulative burnup, CBU, can be represented as

$$CBU(t_n)_j = \sum_{i=1}^{n} BU(d_i)_j \qquad (1)$$

for the j^{th} fuel assembly and

$$C\overline{BU}(t_n) = \sum_{i=1}^{n} \overline{BU}(d_i) \qquad (2)$$

for the average fuel assembly.

Table 1. Depletion schedule for ALRR, inner core ring of fuel assemblies

Step no.	Steplength BU(d) (MWD/MTU)	Cumulative burnup CBU(t_n) (MWD/MTU)
1	505	503
2	1,010	1,510
3	1,020	2,530
4	509	3,040
5	21,200	24,200
6	21,100	45,400
7	21,300	66,700
8	21,400	88,000
9	21,300	109,000
10	21,200	131,000
11	21,100	152,000
12	17,900	170,000

Here

$$t_n = \sum_{i=1}^{n} d_i \tag{3}$$

where

n = the number of burnup steps.

Since $BU(d_i)_j = P_j^i \cdot d_i$ and $P_j^i = RP_j^i \cdot \bar{P}$, the expression for the cumulative burnup becomes

$$CBU(t_n)_j = \sum_{i=1}^{n} P_j^i \cdot d_i$$

$$= \sum_{i=1}^{n} (RP_j^i)(\bar{P})(d_i) \tag{4}$$

where

P_j^i = the power produced by the j^{th} fuel assembly during the i^{th} burnup step

RP_j^i = the relative power of the j^{th} fuel assembly during the i^{th} burnup step.

The average power produced by a fuel assembly is given by

$$\bar{P} = \frac{Q_t}{N_{FA}} = \sum_{j=1}^{N_{FA}} \frac{P_j}{N_{FA}} \tag{5}$$

where

Q_t = the reactor thermal power

N_{FA} = the number of fuel assemblies.

The relative power produced by the j^{th} fuel assembly is given by

$$RP_j = \frac{P_j}{\bar{P}} . \tag{6}$$

Furthermore, the power produced by the j^{th} fuel assembly, P_j, is given by

$$P_j = q'''_j V_{FA} \tag{7}$$

or

$$P_j = G \Sigma_{fj} \bar{\phi}_j V_{FA} \tag{8}$$

where

G = the energy released per fission

q_j''' = the volumetric thermal source strength of the j^{th} fuel assembly

Σ_{fj} = the macroscopic fission cross section for the j^{th} fuel assembly

$\bar{\phi}_j$ = the average flux in the j^{th} fuel assembly

V_{FA} = the volume of a fuel assembly.

On substituting Equation 8 and Equation 5 into Equation 6 one has

$$RP_j = \frac{P_j}{\bar{P}} = \frac{G\Sigma_{fj}\bar{\phi}_j V_{FA}}{\sum_{j=1}^{N_{FA}} \frac{G\Sigma_{fj}\bar{\phi}_j V_{FA}}{N_{FA}}} \tag{9}$$

or

$$RP_j = \frac{N_{FA}\Sigma_{fj}\bar{\phi}_j}{\sum_{j=1}^{N_{FA}} \Sigma_{fj}\bar{\phi}_j} \quad . \tag{10}$$

Finally, substituting Equation 9 into Equation 4 results in the desired expression for the cumulative burnup for the j^{th} fuel assembly

$$CBU(t_n)_j = \sum_{i=1}^{n} \left[\frac{N_{FA} \Sigma_{fj} \bar{\phi}_j}{N_{FA} \sum_{j=1} \Sigma_{fj} \bar{\phi}_j} \right]_j (\bar{P})(d_i) \quad . \qquad (11)$$

IV. RESULTS OF COMPUTER-AIDED STUDIES

Two different computer programs, LEOPARD and FOG, were used in this study to calculate the Σ_{fj} and $\bar{\phi}_j$ needed for application of Equation 11. LEOPARD, an abbreviation for Lifetime Evaluating Operations Pertinent to the Analysis of Reactor Designs, was used to obtain few group macroscopic cross sections from input data such as fine group macroscopic cross sections, fuel plate composition, fuel temperature, cladding temperature, moderator temperature, reactor pressure, etc. These macroscopic cross sections are then used as input data to FOG. FOG is a one-dimensional neutron diffusion code with the capability of calculating one-dimensional neutron flux profiles.

A great many of the parameters utilized in nuclear engineering are functions of the energy of the neutrons involved. Cross sections for various types of nuclear reactions, such as scattering, absorption, and fission, are often very sensitive to neutron energy, especially near resonance peaks. Representing these cross sections as explicit functions of energy is usually impossible. Instead the energy range of interest is divided into a large number of small intervals, and the cross sectional data within each of these "fine mesh" intervals are stored in a "library." Thus, there would be a library for each isotope and for each

reaction of interest. When a macroscopic cross section for a certain isotope and reaction is needed, this fine mesh data can be coalesced for just a few groups, such as one fast group and one thermal group which was done in this study. The above technique is called the multigroup technique.

Obtaining the few group cross sections is just the beginning of the problem. The second major endeavor is to determine the flux profiles. FOG accomplishes this by solving a multigroup diffusion equation of the form:

$$-D^i \nabla^2 \phi^i + \Sigma_T^i \phi^i = \chi^i S(r) + \Sigma_{s,(i-1)\to i} \phi^{(i-1)}. \qquad (12)$$

The symbols are defined as:

ϕ^i = neutron flux in the i^{th} group

D^i = diffusion coefficient in the i^{th} group

Σ_T^i = total removal for the i^{th} group

$$= D^i (B^2)^i + \Sigma_a^i + t^P \Sigma_P^{th} + \Sigma_{s,i\to(i+1)}$$

where

$(B^2)^i$ = transverse buckling for the i^{th} group

Σ_a^i = absorption cross section for the i^{th} group

Σ_P^{th} = poison cross section in the thermal group

t^P = ratio of poison in group i to thermal poison cross section

$\Sigma_{s,i\to j}$ = scattering or transfer coefficient from group i to group j

χ^i = the integral of the fission spectrum over the lethargy range represented by group i

r = distance measured from the origin

$\nabla^2 = d^2/dr^2 + P/r \frac{d\phi}{dr}$ (P = 0 in plane geometry; P = 1 in cylindrical geometry; P = 2 in spherical geometry)

S = normalized source density

$$= \sum_{i=1}^{NOG} \frac{(\nu\Sigma_f)^i \phi^i}{\lambda}$$

where

NOG = number of groups

Σ_f^i = the fission cross section in the i^{th} group

ν^i = the average number of neutrons produced by a fission in the i^{th} group

λ = the eigenvalue, which is related to the multiplication factor K.

These coupled differential equations are then solved by an iterative process.

This was a very general sketch of the multigroup method and the diffusion theory code, FOG. A more detailed description and more complete theoretical basis for the FOG code

can be found in the literature (8).

The purpose of this study was to combine a nonspatial macroscopic generating code, LEOPARD, with a one-dimensional multigroup diffusion theory code, FOG, to deplete the core of the ALRR. As stated previously, these codes were used to calculate the Σ_{fj} and ϕ_j needed as input data for Equation 11 at each burnup step.

One-dimensional codes are best suited for investigating fuel depletion in zonal batches of fuel assemblies. For this purpose a reactor core is assumed to be cylindrical and the computations are carried out along the radial dimension. Axial variations in the power distribution are neglected. To adjust for axial leakage, the transverse buckling, calculated with reflector savings, was used in the code input. The cylindrical core consists of circular regions. Although an individual fuel assembly cannot be accounted for in one-dimensional codes, the zones can be subdivided so that all fuel assemblies within the same radial location are in one annular ring.

Therefore, it was assumed in this study that the ALRR core can be divided into three annular fuel regions of widths 20.00 centimeters, 10.97 centimeters, and 9.66 centimeters. Each core region contains a ring of fuel assemblies (Figure 1). The ALRR fuel assemblies are arranged in circular

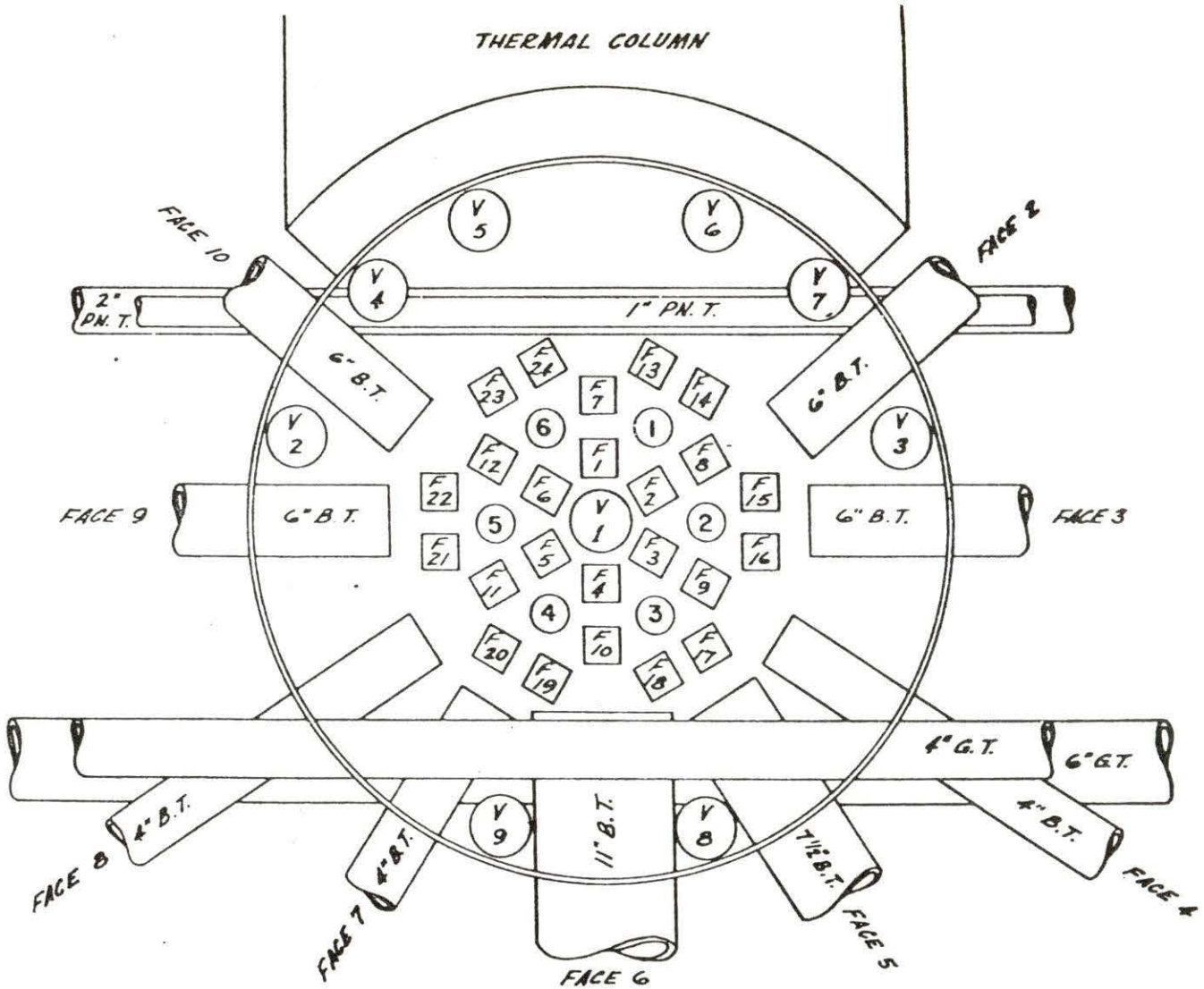

Figure 1. Geometric relation between the core, control rods and some of the experimental thimbles for 24 element core

patterns with their centerlines forming three concentric circles of 13.33 centimeters (6 assemblies), 26.67 centimeters (6 assemblies), and 35.28 centimeters (12 assemblies). In addition to the core region, the ALRR possesses two radial reflectors. The core and radial reflectors are represented in Figure 2. A complete description of the ALRR is given in Reference 6.

In order to obtain the flux profiles in the radial direction throughout the reactor for each depletion step which was needed for application of Equation 11, it was necessary to input two-group constants obtained from the LEOPARD code into the FOG code. The two-group constants for the D_2O and H_2O reflectors of the ALRR were reported by Crudele (9). The macroscopic parameters which were the output data from the LEOPARD code are shown in Table 2.

Good discussions of the various approximations utilized and the neutron physics involved in LEOPARD (10) appear in the 1973 Ph.D. dissertation by Crudele (9) and the Westinghouse report by Strawbridge (11). For purposes of this study attention was focused on the use of LEOPARD rather than its theoretical basis. However, a brief discussion of LEOPARD seems appropriate. The LEOPARD computer program determines fast and thermal spectra and optionally computes fuel depletion for a dimensionless reactor before each burnup step.

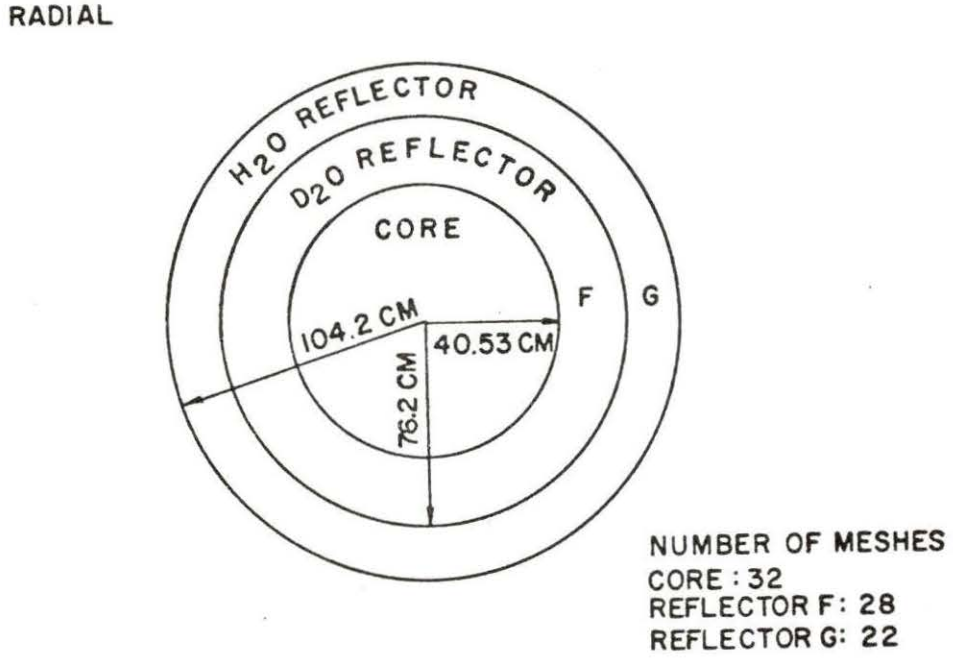

Figure 2. Geometry for FOG calculations

Table 2. Two group neutron cross sections for 24 element ALRR fuel cell

Depletion step (n)	Neutron energy group[a]	D cm	Σ_a cm^{-1}	Σ_R^b cm^{-1}	$\nu\Sigma_f$ cm^{-1}
1	1	1.54	3.27-03	3.69-03	4.68-03
	2	1.07	5.17-02	0.00	1.01-01
2	1	1.54	3.28-03	3.71-03	4.69-03
	2	1.07	5.20-02	0.00	1.01-01
3	1	1.54	3.28-03	3.71-03	4.69-03
	2	1.07	5.22-02	0.00	1.01-01
4	1	1.54	3.28-03	3.71-03	4.69-03
	2	1.07	5.22-02	0.00	1.01-01
5	1	1.54	3.28-03	3.71-03	4.69-03
	2	1.07	5.22-02	0.00	1.01-01
6	1	1.54	3.28-03	3.73-03	4.67-03
	2	1.07	5.22-02	0.00	9.98-02
7	1	1.54	3.28	3.74-03	4.65-03
	2	1.07	5.21-02	0.00	9.93-02
8	1	1.54	3.28-03	3.75-03	4.62-03
	2	1.07	5.19-02	0.00	9.87-02
9	1	1.54	3.28-03	3.75-03	4.60-03
	2	1.07	5.16-02	0.00	9.83-02
10	1	1.54	3.28-03	3.75-03	4.56-03
	2	1.08	5.10-02	0.00	9.81-02
11	1	1.54	3.29-03	3.72-03	4.52-03
	2	1.08	4.98-02	0.00	9.83-02
12	1	1.54	3.30-03	3.67-03	4.47-03
	2	1.08	4.82-02	0.00	9.89-02

[a]Notation 1 refers to fast group, i.e., neutrons of energy above 0.625 eV; 2 refers to neutrons of lower energy.

[b]Notation 3.69-03 equals 3.69×10^{-3}.

This second task is pursued by performing the spectra calculations, calculating fuel depletion for a given time increment, recalculating spectra, etc. This second option was applied for the first time with ALRR data in this study.

For the analysis of plate type fuel elements which are commonly used in highly enriched uranium, heavy water reactors of the ALRR-type, the LEOPARD routine which computes the thermal flux disadvantage factor for a cylindrical unit cell was modified by Kim (1). A method proposed by Bhalla (12) for slab geometry was used. Figure 3 is an illustration of the arrangement of a slab fuel cell. The cell dimensions and composition of a ALRR fuel plate used in this study are given in Table 3.

The two-group constants, D, Σ_a, Σ_R, and $\nu \cdot \Sigma_f$ generated by LEOPARD at each depletion step were then used as input data into the FOG code at each burnup step to obtain the average fluxes $\bar{\phi}_j$ in each region. The averaged results of the FOG calculations for each fuel region ring and depletion step are given in Table 4. Radial variations in the neutron spectrum were considered by using different two-group constants in the core, in the D_2O reflector and in the H_2O reflector. These separate regions are illustrated in Figure 2.

The fast and thermal relative fluxes as calculated by FOG were graphed as functions of reactor radius at various intervals during core life. Figure 4 is a representation of

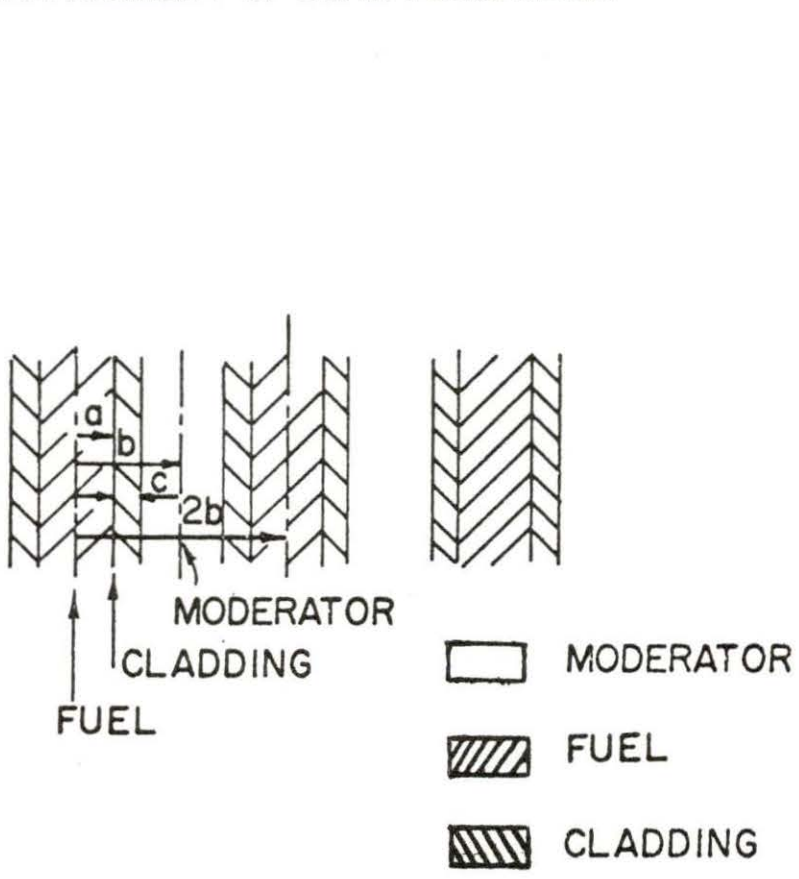

Figure 3. Arrangement of slab fuel cell

Table 3. Cell dimensions and composition of ALRR fuel plate

	Inches	Cm
Fuel length	23.5	59.7
Average fuel width	2.48	6.29
Fuel matrix thickness	0.020	0.051
Volume of fuel matrix	1.17 in^3	19.1 cm^3
Clad thickness	0.020	0.051
Water channel thickness	0.110	0.279
Plate thickness	0.060	0.152
U volume per plate		0.630 cm^3
Al volume per plate		18.5 cm^3
U-235 concentration in fuel	.00147 x 10^{24}	atoms/cm^3
U-238 concentration in fuel	.000106 x 10^{24}	atoms/cm^3
Al concentration in fuel	.0578 x 10^{24}	atoms/cm^3
D_2O in moderator 99.7 mol % D_2O	.0331 x 10^{24} (68°F)	molecules/cm^3
H_2O in moderator	.000100 x 10^{24} (68°F)	molecules/cm^3
D/H in moderator 99.7 mol % D_2O	332	

Table 4. Volume-averaged region fluxes at all depletion steps

Depletion step (n)	Number of energy groups	ϕ_1	ϕ_2	ϕ_3
1	1	0.958	0.855	0.679
	2	0.0688	0.0685	0.120
2	1	0.958	0.854	0.678
	2	0.0688	0.0683	0.119
3	1	0.958	0.854	0.677
	2	0.0675	0.0681	0.119
4	1	0.958	0.854	0.677
	2	0.0675	0.0681	0.119
5	1	0.958	0.854	0.677
	2	0.0675	0.0681	0.119
6	1	0.958	0.854	0.677
	2	0.0678	0.0684	0.119
7	1	0.958	0.853	0.677
	2	0.0681	0.0687	0.119
8	1	0.958	0.853	0.677
	2	0.0685	0.0691	0.120
9	1	0.958	0.854	0.677
	2	0.0697	0.0697	0.120
10	1	0.958	0.855	0.678
	2	0.0698	0.0706	0.122
11	1	0.959	0.857	0.681
	2	0.0710	0.0722	0.124
12	1	0.960	0.861	0.685
	2	0.0727	0.0744	0.128

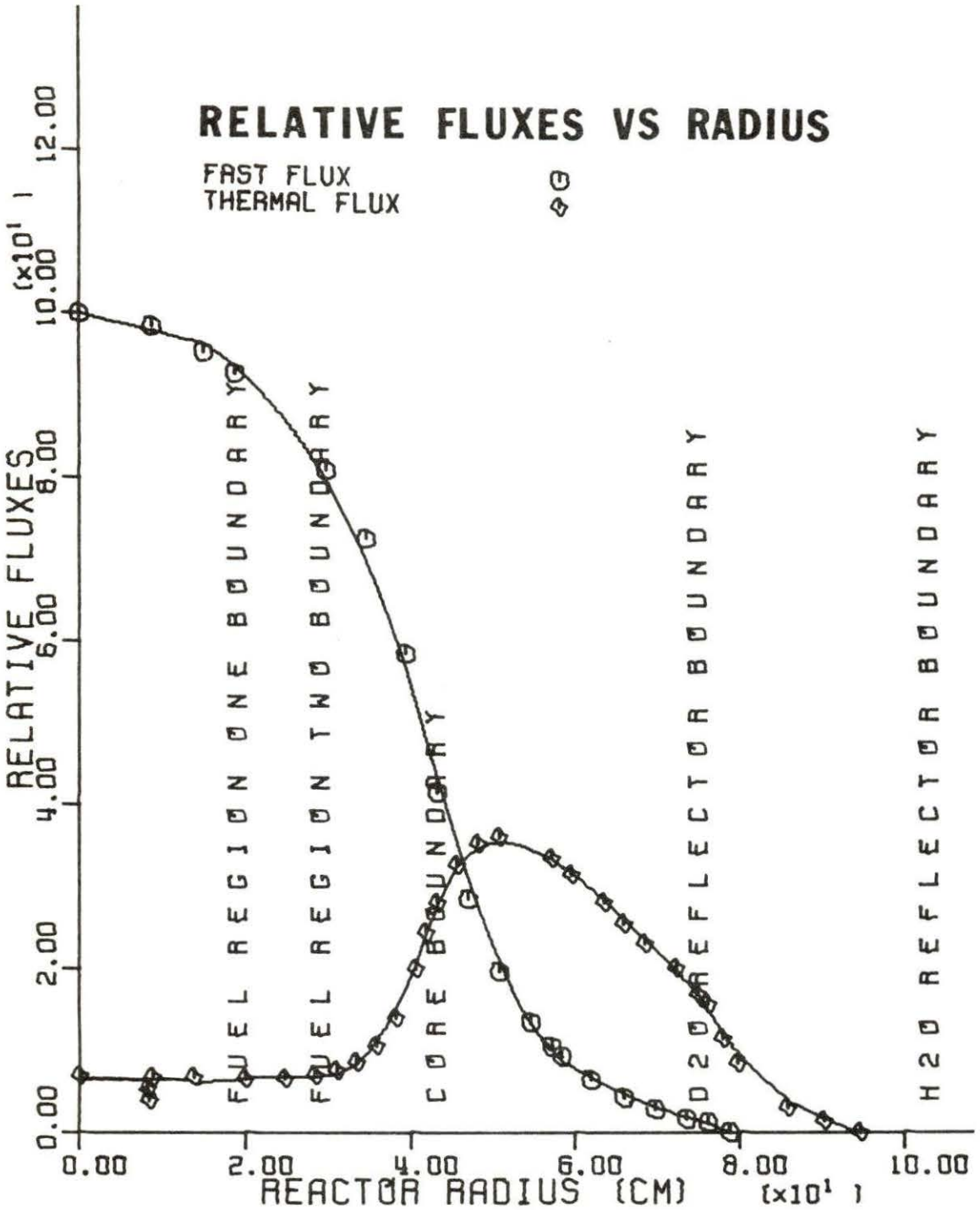

Figure 4. Two group neutron flux profiles at BOL

the radial flux profiles at the beginning of core life (BOL). Midway through core life the flux profiles appear as in Figure 5. Finally, Figure 6 represents the flux profile at the end of core life (EOL). These results indicate that the flux profiles remain approximately constant throughout the core life of the ALRR.

These flux profiles compare favorably with other heavy water research reactor flux profiles found in the literature (5). The thermal flux exhibits a maximum in the D_2O reflector at a short distance from the core-reflector interface. This arises from the fact that in the reflector slow neutrons are produced by the slowing down of fast ones, but they are not absorbed as strongly as in the core. Farther from the interface, the slow neutron flux, like the fast flux, falls to zero.

In applying Equation 11, NFA was set equal to 24, the number of fuel assemblies in the fully loaded ALRR core. Using Equation 10 and data from LEOPARD and FOG calculations, the relative power (RP) for each ring of fuel assemblies was calculated at each of the burnup steps. The results of the relative power calculations can be found in Table 5. It can be seen that the outer ring of fuel assemblies deplete more rapidly since RP > 1. The relative power is approximately 1.15 for the outer ring. The inner ring of fuel assemblies

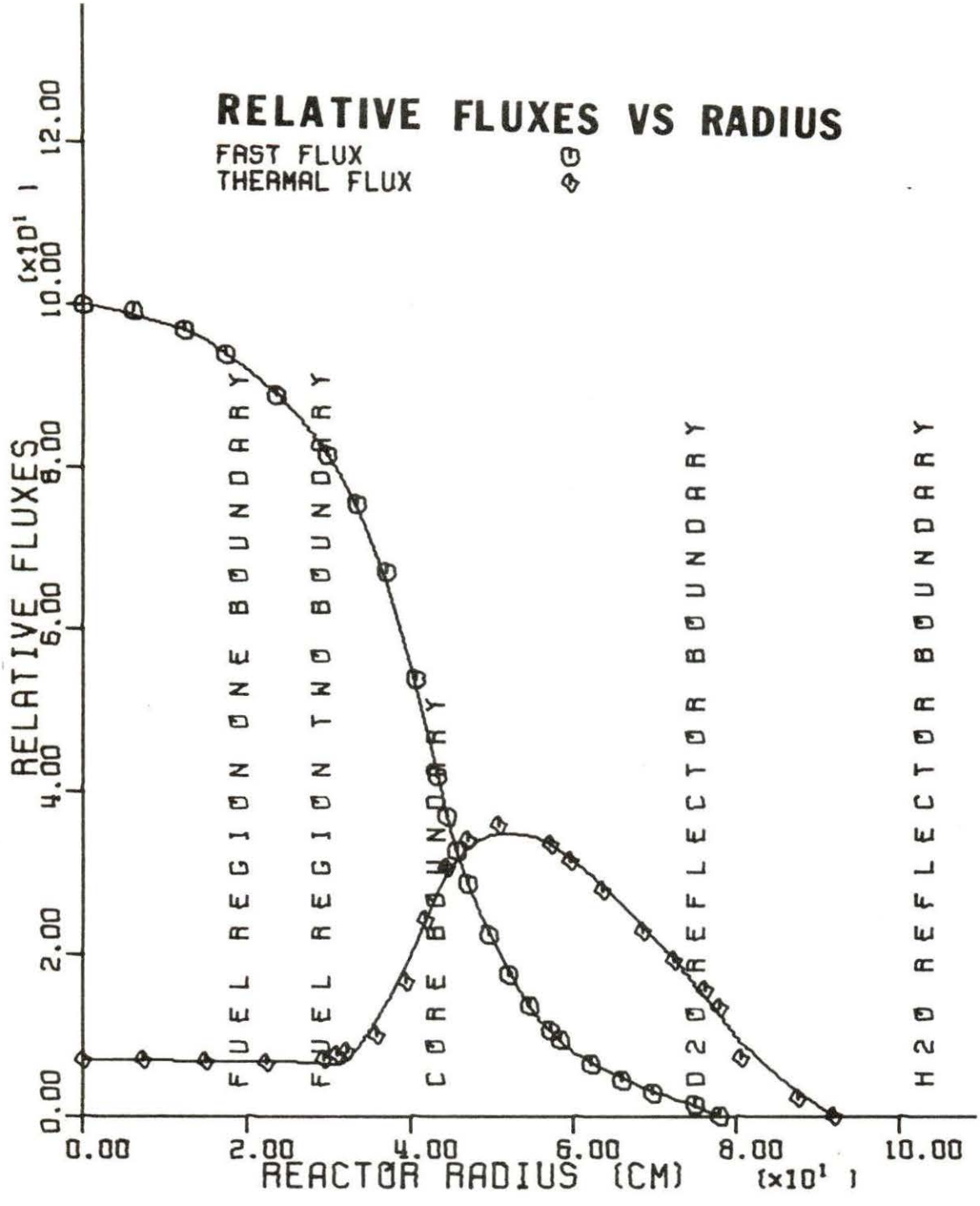

Figure 5. Two group neutron flux profiles at core mid-life

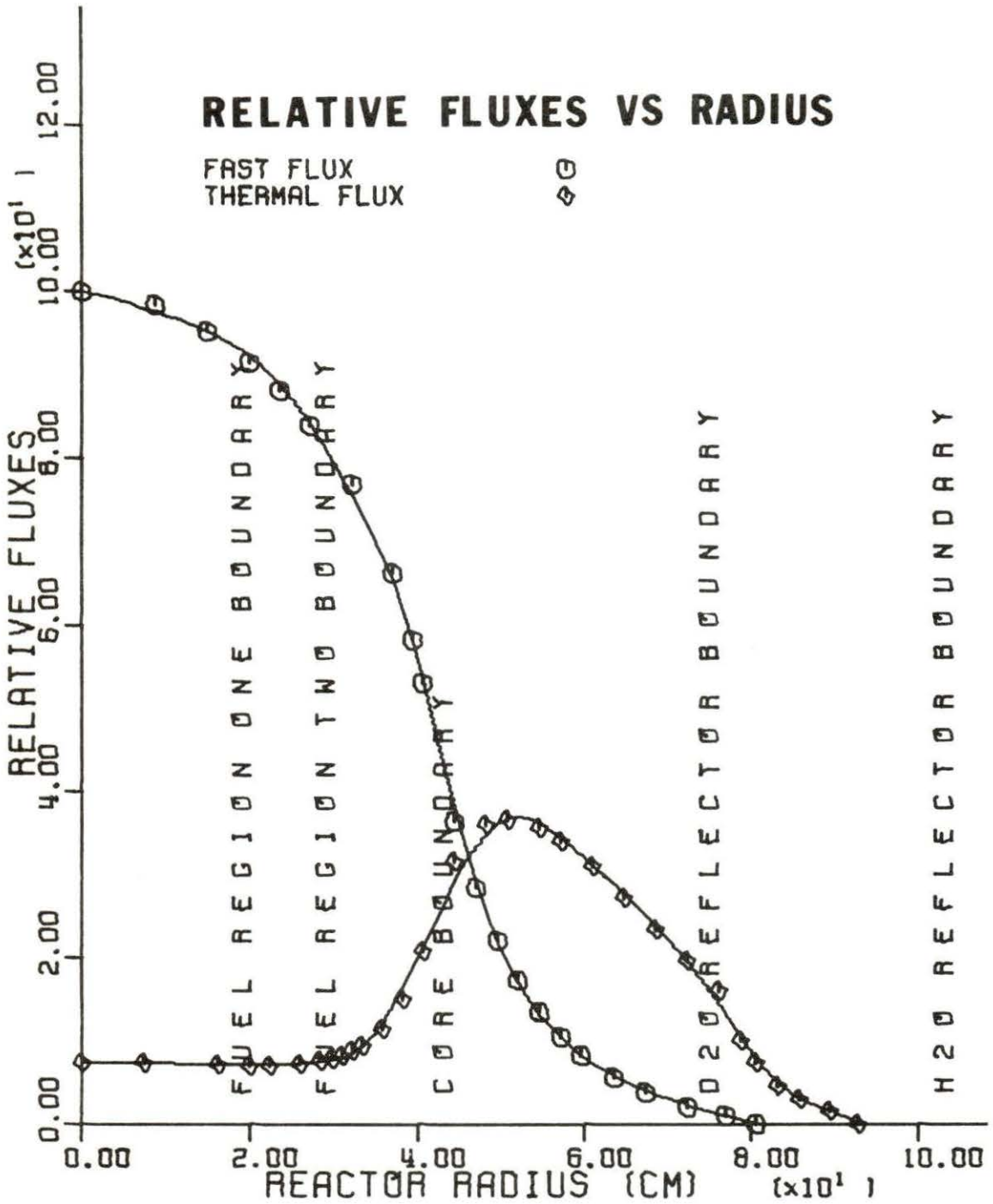

Figure 6. Two group neutron flux profiles at EOL

Table 5. Relative power of fuel assemblies with depletion in three core regions

Depletion step (n)	Region 1 (inner region)	Region 2 (intermediate region)	Region 3 (outer region)
1	0.858	0.828	1.16
2	0.859	0.828	1.16
3	0.861	0.828	1.16
4	0.862	0.830	1.15
5	0.863	0.830	1.15
6	0.864	0.831	1.15
7	0.864	0.831	1.15
8	0.865	0.832	1.15
9	0.870	0.830	1.15
10	0.860	0.830	1.15
11	0.857	0.830	1.16
12	0.849	0.829	1.16

has a relative power of about 0.86 while the center ring has a relative power of approximately 0.83.

To obtain the results from Equation 11 it was necessary to use an effective fission cross section from a two group model. It was assumed that the total fission rate per unit volume, F, is

$$F = \phi_1 \Sigma_1 + \phi_2 \Sigma_2$$

where

ϕ_1 = fast flux

ϕ_2 = thermal flux

Σ_1 = fast fission cross section

Σ_2 = thermal fission cross section.

Factoring out ϕ_2, one obtains

$$F = \left(\Sigma_1 \frac{\phi_1}{\phi_2} + \Sigma_2 \right) \phi_2$$

$$F = \Sigma_{eff} \phi_2$$

where

$$\Sigma_{eff} = \Sigma_1 \frac{\phi_1}{\phi_2} + \Sigma_2.$$

It is assumed that the fast to thermal flux ratio remains constant.

Table 6 presents the effective fission cross sections for each core region and depletion step as calculated from LEOPARD and FOG output data.

For the one-dimensional depletion analysis it was assumed that the ALRR was operated continuously at a constant power of 5.0 megawatts, the reactor's maximum thermal output. It was also assumed that no perturbations in the flux were caused by

Table 6. Effective one group cross sections in each core region

Depletion step (n)	$\Sigma_{eff\ 1}$ cm^{-1}	$\Sigma_{eff\ 2}$ cm^{-1}	$\Sigma_{eff\ 3}$ cm^{-1}
1	0.0690	0.0658	0.0527
2	0.0688	0.0656	0.0524
3	0.0688	0.0656	0.0524
4	0.0687	0.0656	0.0523
5	0.0687	0.0655	0.0523
6	0.0682	0.0650	0.0519
7	0.0677	0.0645	0.0516
8	0.0671	0.0640	0.0513
9	0.0666	0.0635	0.0510
10	0.0661	0.0630	0.0507
11	0.0655	0.0624	0.0505
12	0.0649	0.0619	0.0504

the use of experimental beam ports. The average fuel loading per element was 176.5 grams uranium containing 164.5 grams U-235 for an enrichment of 93.2 per cent. This resulted in a total core loading of 4236.0 grams of uranium. The LEOPARD calculated values of the macroscopic fission cross section and average burnup using this information are shown in Table 7. These values were used in application of Equation 11.

The results of the combination of LEOPARD and FOG computer programs in the one-dimensional depletion analysis

Table 7. Macroscopic fission cross section and nonspatial burnup

Depletion step (n)	Number of energy group	Σ_f (cm^{-1})	Burnup (MWD/MTU)
1	1	.00193	590
	2	.0417	
2	1	.00194	1180
	2	.0414	
3	1	.00194	1180
	2	.0413	
4	1	.00194	590
	2	.0413	
5	1	.00194	24600
	2	.0413	
6	1	.00193	24600
	2	.0409	
7	1	.00192	24600
	2	.0407	
8	1	.00191	24600
	2	.0405	
9	1	.00190	24600
	2	.0403	
10	1	.00188	24600
	2	.0402	
11	1	.00187	24600
	2	.0403	
12	1	.00185	21000
	2	.0405	

are given in Table 8. These results indicate that the outer ring of fuel elements in the ALRR deplete more rapidly than the inner two fuel regions. This is consistent with the variation in the thermal flux across the core. The burnups after approximately 170 days of reactor operation for the inner, intermediate, and outer rings where found to be 18.6, 17.9, and 25.3 atom per cent respectively. This resulted in an averaged core burnup of 21.7 atom per cent. This agrees reasonably well with the determination made by McCorkle and Norman (7) of an average burnup for 21 assemblies of 20.00 per cent and a maximum burnup of 21.93 per cent.

In Table 8 the burnup is expressed in terms of exposure units, MWD/MTU. In order to determine the atom per cent burnup β, the following expression for the exposure E in megawatt days per metric tonne of fuel of atomic weight A was used.

$$E = \beta \frac{\text{atoms fissioned}}{\text{atom of fuel}} \times \frac{192 \text{ MeV liberated}}{\text{fission}}$$

$$\times \frac{6.02 \times 10^{23} \text{ atoms}}{\text{mole}} \times 1.854 \times 10^{-24} \frac{\text{megawatt day}}{\text{Mev}}$$

$$\times \frac{\text{mole of fuel}}{\text{A grams}} \times \frac{10^{6} \text{ grams}}{\text{metric tonne}}$$

or

$$E = \frac{2.143 \times 10^{8}}{A} \beta \frac{\text{MWD}}{\text{MTU}} .$$

Table 8. Summary of burnups by depletion step and region for ALRR

Depletion step (n)	Region 1 (MWD/MTU)	Region 2 (MWD/MTU)	Region 3 (MWD/MTU)
1	505	489	684
2	1,010	977	1,370
3	1,020	977	1,370
4	509	490	679
5	21,200	20,400	28,300
6	21,100	20,400	28,300
7	21,300	20,400	28,300
8	21,400	20,500	28,300
9	21,300	20,400	28,300
10	21,200	20,400	28,300
11	21,100	20,400	28,500
12	17,900	17,400	28,500
Total burnup	170,000	163,000	231,000

For highly enriched uranium such as is used in the ALRR,

$A \triangleq 235$, and

$$E = 9.119 \times 10^{5} \beta \frac{MWD}{MTU}$$

or

$$E = 0.1851 \times 10^{2} \beta \text{ PJ/KG.}$$

V. APPROXIMATIONS USED IN STUDY

The purpose of this study was to get an approximation for the variation in fuel depletion in each ring of fuel assemblies in the ALRR. The purpose of this section is to point out some of the approximations used and how they differ from the actual situation existing in the ALRR. An attempt was made to verify the accuracy of the relative power distribution by changing the U-235 number density in the LEOPARD calculation by 1% and then recalculating the relative power at several depletion steps. It was found that the relative power differed by an average of 0.0066.

1. In this study, regions were defined having constant material and nuclear properties, such as temperatures, diffusion constants, and cross sections. In actuality, many of these parameters would have continuously varying profiles.

2. LEOPARD uses cell-averaging techniques to treat the reactor as being homogeneous, when the actual reactor is composed of discrete fuel assemblies, channel walls and control rods.

3. In this study only the radial dimension of burnup was considered.

4. The use of two neutron energy groups is an approximation to the continuous distribution of neutron energies.

VI. SUMMARY AND CONCLUSIONS

The major objective of this study was to obtain a one-dimensional depletion approximation for the ALRR. It was found that inner two rings of fuel assemblies depleted less rapidly than the outermost third ring of fuel assemblies due to lower relative flux levels and relative power levels throughout the ALRR core life. The core averaged burnup was found to be approximately 21.7 atom per cent which compared favorably with other studies.

It was also found that the thermal flux profile of the ALRR was consistent with other heavy water research reactors of its type and that these flux profiles remained approximately constant throughout the reactor's core life.

The application of the one-dimensional depletion model used in this study has proven to be a simple method to apply. However, its application to the ALRR could not make full use of its potential, due to the small number of fuel assemblies of the ALRR and the slowly varying macroscopic cross sections which made their parameterization unnecessary.

In addition, the information discovered could prove useful in core reload considerations and in fuel management of the ALRR or reactors in general. Also the analysis of the relative power distribution throughout core life verifies the safe power distribution of the ALRR. Furthermore, knowledge

of depletion could lead to possible more efficient use of research reactor fuel.

It should be noted that this study was originated before the decision had been made to shutdown the ALRR.

VII. LITERATURE CITED

1. Kim, Y. S. 1970. Development of design procedures to determine reactor physics characteristics of high-enriched heavy water moderator reactors. NUS-652. (NUS Corporation, Waltham, Mass.)

2. Naughton, W. F., S. H. Levine, and W. F. Witzig. 1972. TRIGA core management model. Trans. Am. Nuc. Soc. 15: 895.

3. Cenko, M. J., S. H. Levine, and W. F. Naughton. 1972. Computer program for TRIGA fuel management. Trans. Am. Nuc. Soc. 15:894-895.

4. Schmid, Loren C. 1971. Critical assemblies and reactor research. John Wiley & Sons, Inc., New York, N.Y.

5. Chastain, J. W., Jr. 1958. U.S. research reactor operation and use. Addison-Wesley Publishing Co., Reading, Mass.

6. AMF Atomics. 1964. Final hazards summary report for Ames Laboratory Research Reactor. (American Machine & Foundry Company, Greenwich, Conn.) 320 pp.

7. McCorkle, W. H. and J. H. Norman. 1967. Fuel burnup, refueling operations and general utilization of the ALRR. (IS-1600, Paper 7, Ames Laboratory, Ames, Ia.) 48 pp.

8. Flatt, H. P. 1961. The FOG one-dimensional neutron diffusion equation codes. NAA-SR-6104. (Atomics International, Canoga Park, Cal.) 47 pp.

9. Crudele, Joseph Samuel. 1973. Study of Ames Laboratory Research Reactor physics characteristics by few group design techniques. Ph.D. Thesis. Iowa State University, Ames, Ia. 119 pp.

10. Barry, R. F. 1963. LEOPARD, a spectrum dependent non-spatial depletion code for the IBM-7094. WCAP-3269-26. (Westinghouse Atomic Power Division, Pittsburgh, Pa.) 65 pp.

11. Strawbridge, L. E. 1963. Calculation of lattice parameters and criticality for uniform water moderated lattices. WCAP-3269-25. (Westinghouse Atomic Power Division, Pittsburgh, Pa.) 33 pp.

12. Bhalla, C. P. 1962. Thermal flux disadvantage factor for slab geometry. Nuc. Sci. Eng. 12:311.

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