DPST-67-83-9

# USAEC - AECL COOPERATIVE PROGRAM

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Savannah River Laboratory
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# SECTION I

## REACTOR PHYSICS EXPERIMENTS

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### INTRODUCTION

One of the Savannah River Laboratory (SRL) projects under the Cooperative Program for this fiscal year is the extension of the multigroup one-dimensional HAMMER code for reactor cell calculations to two dimensions by incorporating a collision probability treatment for each energy group. As mentioned in DPST-67-83-8, the possibility of accomplishing this end by combining the one-group collision probability CLUCOP code with the HAMMER code was tested for reactor fuel clusters of the CANDU type by using preliminary HAMMER calculations to define energy averaged one-group cross sections and then using these cross sections in CLUCOP calculations. The results were compared with earlier SRL experimental measurements on rod clusters and with thermal activation ratios predicted by the cylindricized HAMMER code alone. Some of the results of these comparisons are given in this report.

### SUMMARY

The CLUCOP code when used in conjunction with HAMMER cross sections gave generally good agreement with measured thermal neutron flux distributions for fuel rod clusters using $D_2O$ or air coolant but only fair agreement for clusters using "HB-40"* organic coolant. The latter effect is believed to be due largely to the inadequacy of the one-group treatment.

Based on these results it appears that the incorporation of the CLUCOP treatment into HAMMER code with its 30 thermal neutron energy groups will give reliable results for cluster calculations.

### DISCUSSION

#### CLUCOP Treatment of Rod Clusters

The CLUCOP(1) code which was developed by A. B. Atomenergi in Sweden and rewritten for the IBM 360 by Combustion Engineering

* Product of Monsanto Co., St. Louis, Mo.

is a one-group, two-dimensional transport theory code which assumes isotropic neutron scattering for reactor cell calculations. The code first calculates a neutron probability matrix from the cell geometry. Flux distributions within a cell are computed by solving the linear system of equations:

$$V_i \Sigma_i \overline{\Phi}_i = \sum_j V_j (S_j + \Sigma_{sj} \overline{\Phi}_j) \; P_{i=j}$$

where the V's are volumes, the Σ's macroscopic cross sections, the ϕ's fluxes, the S's sources, and the P's collision probabilities.

Input to the code consists of a description of the cell geometry along with the neutron cross sections and sources for each region. The output includes the neutron flux and absorptions in each region and the collision probability matrix.

The major advantage which CLUCOP offers is the ability to describe exactly clusters of fuel rods or other materials through two-dimensional geometry. This, of course, obviates the necessity for the construction of a "ring model" or other one-dimensional approximation of the cluster as is necessary for a HAMMER[2] treatment. It is also possible with CLUCOP to treat both symmetric and asymmetric clusters.

The major disadvantage of the code is that it is monoenergetic. In order to obtain meaningful results from CLUCOP in a calculation of the thermal neutron distribution in a reactor fuel cell, it is necessary to use appropriate average thermal cross sections for the various regions. In the current calculations these average cross sections for the fuel and "HB-40" were obtained from HAMMER runs for a homogenized or ring model of each lattice cell. Nominal values were assumed for all other absorption cross sections.

Scattering cross sections were transport corrected. Nominal values were used for the fuel and cladding and "HB-40" coolant since the flux in the fuel was found to be fairly insensitive to these values. The flux level in the $D_2O$ moderator was found to be quite sensitive to the scattering cross sections for that region, although the flux levels in the fuel were still relatively independent of this parameter. In order to obtain a reasonable value for the moderator cross section, an average over the HAMMER spectrum in the moderator was performed as:

$$\frac{1}{\sigma_{tr}} = \frac{\phi(v)dv}{(\sigma_{so} - \sigma_{si})}$$

The above averaging was done at two points in the moderator of one problem and for an analytic Maxwellian spectrum. The results were:

Inner moderator $\Sigma_{tr}$ = 0.3787 $cm^{-1}$

Outer moderator $\Sigma_{tr}$ = 0.3938

Analytic Maxwellian $\Sigma_{tr}$ = 0.3941

The value which was used in the calculation for all lattices was $\Sigma_{tr} = 0.392\ cm^{-1}$.

The neutron source which must be specified for each region was assumed to be flat in the moderator and zero in the fuel for all lattices. The source in the $D_2O$ and "HB-40" coolants was assumed flat and equal to the source in the moderator.

The foil activations calculated with CLUCOP and HAMMER along with measured foil activations are given in Table I for 19- and 31-rod clusters of natural uranium oxide rods. The dimensions of the clusters and the location of the foils are given in Figure 1. Details of the measurements are given in References 3 and 4. A description of the ring model used for the HAMMER calculations is given in Reference 4.

## REFERENCES

(1) A. Åhlin, Instructions for the Use of the CLUCOP Programme, RFN-202, A. B. Atomenergi, Stockholm, Sweden (1965).

(2) J. E. Suich and H. C. Honeck, The HAMMER System, Heterogeneous Analysis by Multigroup Methods of Exponentials and Reactors, USAEC Report DP-1064, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, S. C. (1967).

(3) H. S. Hilborn (Compiler), USAEC-AECL Cooperative Program, Monthly Progress Report, April 1967, DPST-67-83-4, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, S. C. (1967).

(4) N. P. Baumann, J. L. Crandall, R. L. Olson, G. F. O'Neill, D. J. Pellarin, and V. D. Vandervelde, Lattice Experiments with Simulated Burned-Up Fuel for $D_2O$ Power Reactors, USAEC Report DP-1122, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, S. C. (to be issued).

TABLE I

MEASURED AND CALCULATED FOIL ACTIVATIONS IN ROD CLUSTERS

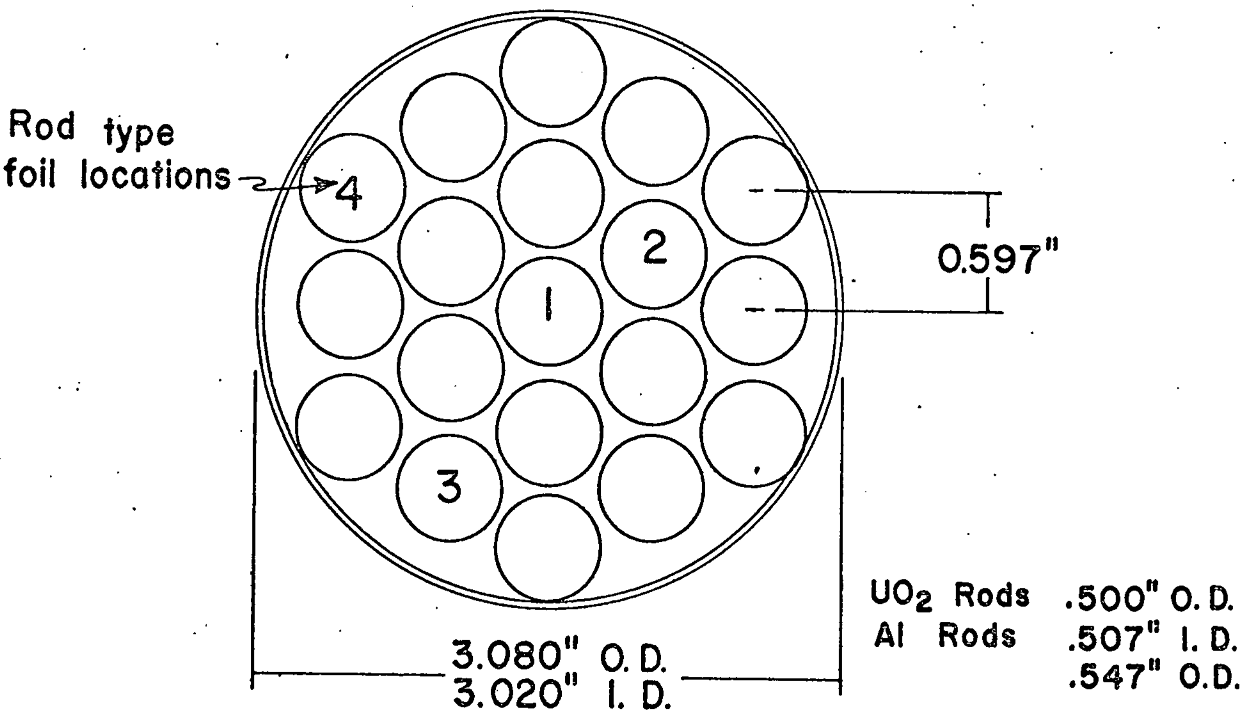
| No. of Rods, Coolant | Rod | Relative Activation: Foil[a] | CLUCOP | HAMMER |
|---|---|---|---|---|
| 19-rod, $D_2O$ | 1 | 1.000 | 1.000 | 1.000 |
| | 2 | 1.072 | 1.068 | 1.084 |
| | 3 | 1.242 | 1.242 | 1.306[b] |
| | 4 | 1.337 | 1.308 | |
| 19-rod, Air | 1 | 1.000 | 1.000 | 1.000 |
| | 2 | 1.063 | 1.056 | 1.074 |
| | 3 | 1.206 | 1.206 | 1.253 |
| | 4 | 1.256 | 1.257 | |
| 19-rod, "HB-40" | 1 | 1.000 | 1.000 | 1.000 |
| | 2 | 1.112 | 1.102 | 1.101 |
| | 3 | 1.401 | 1.369 | 1.433 |
| | 4 | 1.552 | 1.519 | |
| 31-rod, $D_2O$ | 1 | 1.000 | 1.000 | 1.000 |
| | 2 | 1.038 | 1.061 | 1.074 |
| | 3 | 1.170 | 1.202 | 1.249 |
| | 4 | 1.269 | 1.270 | |
| | 5,6 | 1.505 | 1.518 | 1.556 |
| 31-rod, "HB-40" | 1 | 1.000 | 1.000 | 1.000 |
| | 2 | 1.096 | 1.105 | 1.079 |
| | 3 | 1.302 | 1.356 | 1.325 |
| | 4 | 1.420 | 1.481 | |
| | 5,6 | 1.865 | 1.986 | 1.785 |

(a) Average of activations of $^{55}Mn$, $^{186}W$, and $^{63}Cu$ foils.

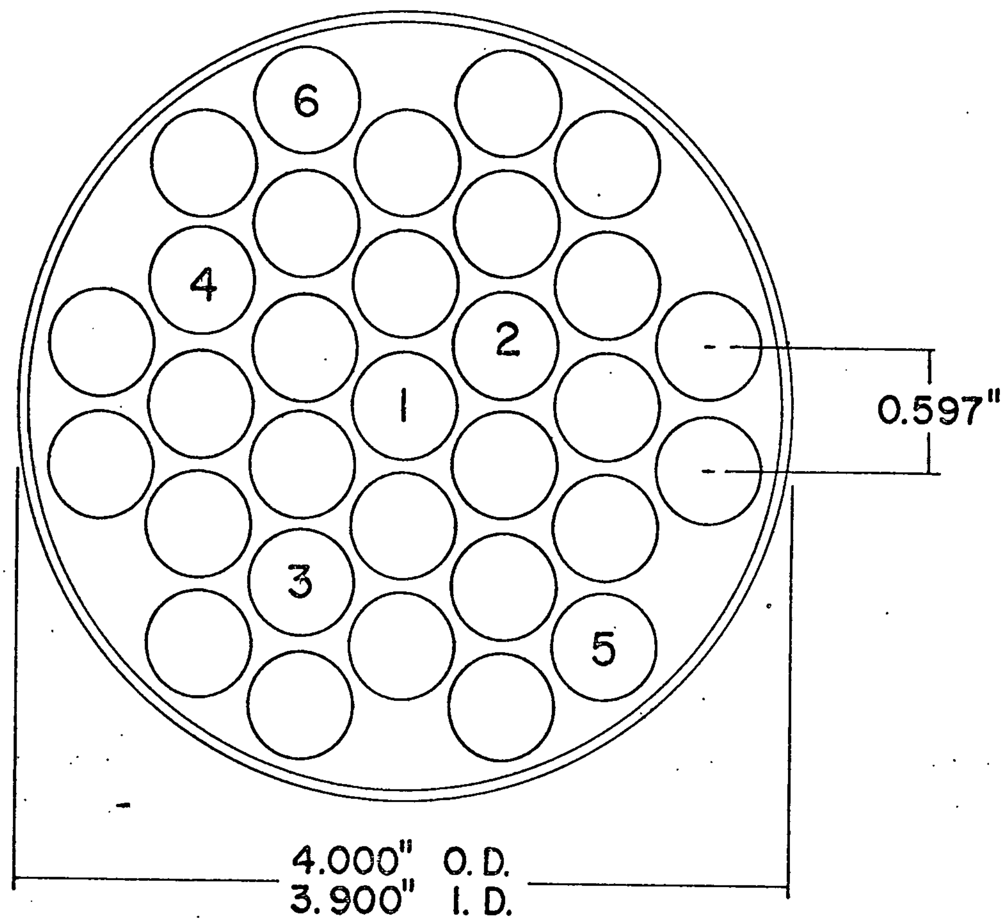
(b) In HAMMER, rods 3 and 4 were combined into one ring.

## Figure 1
## $UO_2$ ROD CLUSTER

### 19 ROD CLUSTER




### 31 ROD CLUSTER



# SECTION II

## AECL IN-CORE FLUX MONITORS

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Reactor Technology Section
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An irradiation test of in-core flux monitors was made in one of the Savannah River Plant reactors to determine the life characteristics of a selection of flux detectors and of the mineral insulation used in their construction. Self-powered flux detectors are relatively new; therefore, confidence in their use hinges to a great extent on proven performance at large integrated exposures. The chief points of interest are 1) integrity of the conductors and sheath during life, 2) life of insulation, and 3) sensitivity.

Irradiation testing of the AECL neutron detector rod described in DPST-66-83-5 has been completed. A report[1] has been written on the irradiation test.

(1) R. F. Byars, AECL In-Core Flux Monitors, DPSPU-67-30-5 E. I. du Pont de Nemours and Co., Savannah River Plant, Aiken, S. C. (1967).

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