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RIBOT: A PHYSICAL MODEL
FOR LIGHT-WATER LATTICES CALCULATIONS

Paolo Loizzo

FEBRUARY 1968

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RIBOT: A PHYSICAL MODEL FOR LIGHT-WATER LATTICES CALCULATIONS

By

Paolo Loizzo*

NUCLEAR EXPERIMENTS AND ANALYSES SECTION
REACTOR PHYSICS DEPARTMENT

February 1968

PACIFIC NORTHWEST LABORATORY
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FOREWORD

The bulk of the work described in this report has been done by the author in cooperation with G. Buffoni and S. Lopez. Most of this work has been published either in Italian or in not readily available literature. The RIBOT code which was originally written in Fortran-II has been converted to Fortran-IV and is operational on the UNIVAC-1108 computer. Thus, to increase the usefulness of the code it seems worthwhile to write here the main physical assumptions of the nuclear model and the input-output instructions for the RIBOT code. Results of an analysis of some recent $\text{UO}_2\text{-PuO}_2\text{-H}_2\text{O}$ lattice experiments using this code are also given.

ABSTRACT

The computer program, RIBOT-5, is a zero dimensional reactor physics code used for calculating criticality and burnup of light-water reactors. The code requires as input, essential extensive reactor parameters (fuel rod radius, water density, etc.).

The nuclear constants (intensive parameters) are calculated with a five-group model (2 thermal and 3 fast groups). A fitting procedure, imitated from the classical engineering approach to problem solving, is systematically employed to reduce the computation time of the code. Zero dimension burn-up calculations are made in an automatic way.

A summary physical description, a check with the latest set of experiments with $\text{PuO}_2\text{-UO}_2$ rods, and a description of the input-output procedures are given in this report.

I. INTRODUCTION

Ribot was a French painter of the impressionist school. A famous Italian horse was named after him in the late fifties. This horse has been very successful such that in the last races it was given 1 to 5 odds by bookmakers. The code RIBOT, is a zero dimension burn-up code for pressurized and boiling water reactors, which couples reliability and speed. Its reliability comes from an extensive check against many reactor experiments on regular lattices and power reactors.⁽¹⁾ It is a direct consequence of the fact that the physical phenomena are basically calculated by standard methods such as codes TEMPEST, MUFT, etc., and often recourse is made to MONTE CARLO methods (like in the calculation of the ^{240}Pu resonance integral). Its speed is a consequence of the fact that a very systematic fitting procedure has been used for many parameters which require complicated calculations. This procedure has been imitated from the classical engineering method of solving problems. For example, the calculation of the thermal cross sections averaged on a Wigner-Wilkins spectrum (usually calculated with the code TEMPEST or similar) has been reduced to a polynomial correlation in function of 3 variables.

The code RIBOT is used as a part of the 2D codes BOLERO⁽¹⁾ and BURNY.⁽¹²⁾ The zero dimension code has been written for use as a survey tool and for checking clean lattice experiments. The reader is invited to read Ref. (1) for the detailed formulae and for the first set of checks against experimental results.

In this paper only a summary physical description and the input-output procedure will be given for the code RIBOT. This code is also

being coupled with a thermal hydraulic design (REPP)⁽²⁾ and an economics code (QUICK)⁽³⁾ to form the code package which will be used for reactor studies.

II. PHYSICAL DESCRIPTION OF THE MODEL

The general model of RIBOT is a five-group model with three fast groups and two (nonoverlapping) thermal groups. The group boundaries are set at 10 MeV, 183 keV, 5.53 keV, 0.625 eV and 0.2 eV.

The group structure for the three fast groups has been chosen in such a way that the removal (scatter-transfer)^{*} and the transport microscopic cross sections can be considered constant within each group and independent from the reactor composition.

The first group is essentially a fast fission and leakage group. More emphasis is given to the latter effect, since the lower boundary of this group is set to the energy where the hydrogen total cross section starts decreasing from the epithermal constant value of about 20 barns. The frequently used cutoff energy of 0.841 MeV gives a better description of the fast-fission effect, but does not allow the use of a constant hydrogen transport cross section for the second group. Our choice of a cutoff at 0.183 MeV, of course, implies that the fast-fission effect is not very precisely described with constant cross sections. The error involved in this approximation is, however, never greater than a few mk. Only absorption related to fast fissions are included in the absorption term for this group.

*NOTE: The removal cross section is not the sum of absorption and scatter transfer cross sections.

The second group is essentially a diffusion group and is treated in the same way as group 1. The absorption terms for groups 2 and 3 are considered to occur in group 3 only. In group 3 the many modes of absorption and fission used are:

1. Infinite dilution resonance integral are used for the light nuclides ($A < 30$).
2. Effective resonance integral representations using semi empirical formulae for ^{238}U , ^{232}Th , and ^{240}Pu which depend on the rod geometry and lattice effects.
3. Resonance absorption for all other $A > 90$ nuclides is treated with NRIA homogeneous formula, but independent from the geometry.

The cross sections used are given in Tables 1 to 4. The ^{238}U and ^{232}Th resonance integrals, are deduced from the Hellstrand's and Pettus' formulae; surface and volume terms are set according to the Forti⁽⁴⁾ formulation.

The scheme for computing the ^{240}Pu resonance integral has been deduced from heterogeneous Monte Carlo calculations.⁽⁵⁾ For small concentrations of ^{240}Pu a smooth transition curve was used from the infinite dilution resonance integral (8400 b) to the actual heterogeneous value for high ^{240}Pu concentrations.

The formulae used for these heterogeneous resonance integral calculations are:

^{238}U

$$I_{\text{eff}} = 8.78 [1 + 0.0053 (\sqrt{T_{\text{Fuel}}} - \sqrt{300})] \\ + D_{\text{eff}} \times (-4.32 + 26.6 \sqrt{S/M}) \times [1 + 0.0125 \sqrt{S/M} (\sqrt{T_{\text{Fuel}}} - \sqrt{300})]$$

and

$$P_{28} = \exp (-N_{28} I_{\text{eff}} / \xi \Sigma_s).$$

^{232}Th

$$I_{\text{eff}} = [8 + D_{\text{eff}} \times (-10.84 + 26.9 \sqrt{S/M})] \times (1 + 0.17 \lg \frac{T_F}{293})$$

^{240}Pu

$$I_{\text{eff}} = 206 [1 - 0.015 (\sqrt{T_{\text{Fuel}}} - \sqrt{300})] \\ + 206 \times D_{\text{eff}} \times \sqrt{S/M_{40}} \times [1 + 0.0075 (\sqrt{T_{\text{Fuel}}} - \sqrt{300})] + 100 \\ (60 < S/M_{40} < 400)$$

or

$$I_{\text{eff}} = 8,400 - 76,000 / (D_{\text{eff}} \times \sqrt{S/M_{40}}) + 100 \quad (S/M > 400)$$

and

$$p_{40} = 1 - \frac{I_{\text{eff}} N_{40}}{\xi \Sigma_s}.$$

We will assume that the reader is familiar with the notation used in these formulae, however they are explained fully in Reference 1.

In the third group, as in the first two groups, the removal cross section, Σ_r , is assumed to be independent from the absorption cross section. This approximation tends to over estimate Σ_r and Σ_a for group 3 (with respect to MUFT), but does not change their ratio.

The thermal constants are calculated in an original way which couples accuracy and speed. The thermal zone is divided into two groups (cutoff point at 0.2 eV) and in each group average cross sections are calculated with the Wigner-Wilkins model. With these constants monoenergetic disadvantage factors are calculated for each group (Amouyal-Benoist method) and the spectrum calculation is done again weighting the nuclide numbers with the disadvantage factors. The originality of the method consists

mainly in the fact that the thermal spectrum is not calculated each time. The calculations were done only once, for appropriate ranges of nuclides ratios ($1/v$ barns/H, $^{235}\text{U}/\text{H}$ and $^{239}\text{Pu}/\text{H}$) and of temperatures, and the cross sections were fitted with polynomial formulae. The absorption, fission, up-scattering and down-scattering cross sections are all given by the fitting. The 2200 m/sec cross sections used in the code are given in Tables I to IV. The calculation of k_{eff} is made assuming the usual point reactor model where the leakage is described by the buckling.

The code also collapses the three fast groups and/or the two thermal groups to yield four (3 fast and 1 thermal), three (1 fast and 2 thermal), and two group (1 fast and 1 thermal) constants.

TABLE I

Miscellaneous Material Cross Sections,
2200 m/sec Constants and Infinite
Dilution Resonance Integrals

<u>Nucleus</u>	<u>$\sigma(2200)$</u>	<u>I_{∞}</u>
H	0.332	0.132
B	761	306
O		
Al	0.24	0.18
Fe	2.62	3.1
Cr	3.1	1.9
Ni	4.6	4
Mn	13.2	13
Zr	0.185	3

TABLE II

Burnable Poison Cross Sections

<u>Nucleus</u>	<u>$\sigma(2200)$</u>	<u>I_{∞}</u>
Sm 149	(1)	0
Sm 150	100	270
Sm 151	12,400	9,300
Sm 152	220	3,000
Eu 151	(1)	5,500
Eu 152	6,000	2,500
Eu 153	450	1,500
Eu 154	1,500	600
Eu 155	14,000	5,000

TABLE III

Pseudo Fission Products' Yield and Cross Sections

<u>U-233</u>			<u>U-235</u>			<u>Pu-239, Pu-241</u>		
<u>y</u>	<u>$\sigma(2200)$</u>	<u>I_{∞}</u>	<u>y</u>	<u>$\sigma(2200)$</u>	<u>I_{∞}</u>	<u>y</u>	<u>$\sigma(2200)$</u>	<u>I_{∞}</u>
#1 0.75	11	67	0.72	13.7	81	0.631	15.8	91
#2 0.25	140	533	0.28	133	560	0.369	130	580
#3 -	-	-	-	-	-	$\hat{8}/\sigma$	300	10,000

TABLE IV

Heavy Nuclides Cross Sections

<u>Nucleus</u>	<u>$\sigma(2200)$</u>	<u>I (3)</u>	<u>Res.</u>	<u>Resonance Treatment</u>	<u>v</u>
Th 232	7.56	-	-	Heterogeneous See Text	-
Pa 233	60	624	0	Homogeneous	-
U 233 (a)	(1)	853	9	Homogeneous	-
U 233 (f)	(1)	1064.1	9	Homogeneous	2.51
U 234	105	723	1	Homogeneous	-
U 235 (a)	(1)	425	11	Homogeneous	-
U 235 (f)	(1)	280	11	Homogeneous	2.45
U 236	6	358	1	Homogeneous	-
U 238	2.71	-	-	Heterogeneous See Text	-
Pu 239 (a)	(2)	522	18	Homogeneous	-
Pu 239 (f)	(2)	293	18	Homogeneous	2.898
Pu 240	(1)	(8400)	-	Heterogeneous See Text	-
Pu 241 (a)	(4)	584	1	Homogeneous	-
Pu 241 (f)	(1)	402	1	Homogeneous	3.06
Pu 242	30	1980	1	Homogeneous	-

Notes to Tables I, II, III, IV

- (1) Non $1/v$ behavior - BNL 325, 2nd Edition or Supplement
- (2) Non $1/v$ behavior - Schmidt's compilation KFK 120
- (3) Where a resonance structure is used, I_{∞} is shown
- (4) Deduced from σ_f assuming $\alpha = 0.344$.

III. COMPARISON WITH EXPERIMENTS

A. Criticality

A very complete comparison has been made with uniform H_2O lattices with enriched UO_2 , ThO_2-UO_2 , PuO_2-UO_2 experiments and with results of PWR and BWR power reactor experiments.⁽¹⁾ Here the latest work will be described, namely the analysis of all PuO_2-UO_2 critical and sub-critical experiments. They are:

- I) The critical mass experiment done at WREC⁽⁶⁾ with 2% PuO_2-UO_2 rods and an isotopic concentration of ^{240}Pu of 8% and 24% (in the total Pu);
- II) Saxton criticals (6.6 wt.% PuO_2-UO_2) done at WREC.⁽⁷⁾ Rectangular critical mass experiments were considered;
- III) EBWR (1.5 wt.% PuO_2-UO_2) approach-to-criticals done at PNL.⁽⁸⁾ The extrapolated critical mass and the exponential B^2 measurements were analyzed;
- IV) Two percent PuO_2-UO_2 experiments in Critical Approach Facility at PNL;⁽⁹⁾ the rods are the same as in (I)
- V) Two percent PuO_2-UO_2 criticals and temperature coefficient measurements in the PRCF at PNL.

The calculated results are given in Table V. The values of k_{eff} in Column 6 of the table were obtained using "experimental" bucklings. The original RIBOT 5 cross section library had a lower value for ν^{239} (2.869) and a higher value of ^{239}I (338 b).

A one dimensional diffusion theory calculation was made using the code EXTERMINATOR 2 in 1-D radial geometry for experiments (I) (II), (IV) and a 2-D (xy) calculation was made for the Saxton

experiments (II) using the RIBOT 5 constants. These results are given in Column 7 of Table V. The difference in k_{eff} between a point reactor calculation and a one or two dimensional calculation of the leakage is obtained by comparing k_{effs} of Columns 6 and 7.

In the EXTERMINATOR calculations five groups were used, although comparisons with 4-group calculations did not show much difference (differences of the order of a few mk). Since the mesh size was quite coarse (1 to 2 cm) a slight overestimate of k_{eff} is expected.

The "theoretical" B^2 was also calculated using the values of k_{eff} obtained using EXTERMINATOR (EXT) and the condensed RIBOT two-group constants in the formula:

$$k_{\text{eff}}(\text{EXT}) = \frac{\nu \Sigma_f^1 + \frac{\nu \Sigma_f^2 + \Sigma_r}{\Sigma_a^2 + D_2 B^2}}{\Sigma_r + \Sigma_a^1 + D_1 B^2} .$$

The resulting B^2 values are shown in the 8th column of Table V. The results of Table V show a disagreement between experiment and calculations. The trends shown in the calculated values of k_{eff} may be interpreted in the following way:

- a) k_{eff} of the WREC 2% lattices, (I) tends to decrease as the moderator-to-fuel (oxide) volume ratio, M/F increases.
- b) The eigenvalues* of the Saxton criticals, (II) do not show a trend with M/F; nevertheless they are higher than the corresponding values for the 2% criticals, (I).

* Eigenvalue meaning k_{eff} .

- c) The eigenvalues of the 2% critical approach experiments (hexagonal lattices), (IV) are 1% lower than the corresponding WREC criticals (square lattices), (I).
- d) At the same M/F in general, the eigenvalues of the 2% experiments increases as the content of ^{240}Pu increases (see, for example exp. WREC 3 and 10, WREC 4 and 11, CAF 1 and 8).

The points (a) and (b) suggest that the error shown is proportional to the concentration of ^{239}Pu in the core. Possible reasons for these errors are:

- I. The shape of η^{239} vs. energy in the thermal zone of the KFK 120 library may not be adequate; in fact, the decrease of η^{239} with energy in the ENDF/B evaluation⁽¹¹⁾ is more prominent; however, it had been previously found that the variation of k_{eff} is not very sensitive to the shape of η^{239} versus energy.*
- II. The epithermal fission resonance integral of ^{239}Pu is too high. Literature values of the fission resonance integral are found as high as 360 b and as low as 290 b. Thus, the sensitivity of k_{eff} to this quantity was investigated in detail.

Adjustment of Cross Sections

The RIBOT calculations were repeated with various values of the fission resonance integral [$I_{\infty f}(239)$] between 278 b to 338 b. Also the value of ν^{239} was changed to get an average value of $k_{\text{eff}} = 1$ for 13 of the critical experiments (the experiments with $M/F > 5$ were excluded from the average). The results are shown in Table VI.

* The η value at 2200 m/sec in KFK 120 (2,079) is very different from the value given by Westcott (2-11). Nevertheless, the decrease of η vs. energy in KFK 120 is much less prominent than in the ENDF/B Library (Leonard shape), in such a way that, at 0.05 eV, where the spectrum of H_2O lattices has its maximum, the two values coincide. Since changing the Pu enrichment does not change the spectrum much, the variation of k_{eff} with the σ shape is not significant for the different H_2O lattices.

TABLE V

Experiments' Analysis

						Calculated Values				
		<u>%240</u>	<u>VF</u>	<u>Boron</u> <u>(ppm)</u>	<u>Exp. B²</u> <u>(M⁻²)</u>	<u>k_{eff}</u> <u>(B² exp.)</u>	<u>k_{eff}</u> <u>(EXT.)</u>	<u>B²</u> <u>(EXT.)</u>	<u>k_{eff}</u> <u>(Final)</u>	
I	WREC	1	8	1.13	0	69.1	1.0193	1.0051	73.3	1.0042
		2	8	1.56		90.1	1.0078	0.9978	93.4	0.9991
		3	8	3.52		106.0	1.0047	0.9890	112.0	0.9935
		4	8	4.37		98.4	1.0044	0.9887	104.4	0.9936
		5	8	8.26		50.3	0.9828	0.9790	51.6	0.9848
		6	8	1.13	261	62.6	1.0229	1.0086	66.8	1.0074
		7	8	3.52	261	83.7	0.9976	0.9914	86	0.9955
		8	8	1.13	526	58.3	1.0196	1.0079	61.6	1.0065
		9	8	3.52	526	63.1	0.9927	0.9914	63.6	-
		10	24	3.52	0	79.5	1.0089	1.0013	82.1	1.0056
		11	24	4.37	0	73.3	1.0042	1.0023	73.9	1.0070
II	Saxton	1	8	1.68	0	108.8	1.0109	1.0039	111.4	0.9958
		2	8	2.17	0	121.5	1.0207	1.0042	128.2	0.9993
		3	8	4.7	0	159.6	1.0042	1.0008	160.3	1.0046
		4	8	5.67	0	159.3	1.0040	0.9983	161.7	1.0016
		5	8	10.73	0	128.4	0.9954	0.9893	131.1	0.9943
		6	8	2.17	337	112.3	1.0244	1.0013	120.2	0.9989
III	EBWR	1	8	1.1	0	48	0.9963	0.9923	40.1	0.9933
		2	8	1.56		65.1	0.9953	0.9824	69.2	0.9854
		3		2.71		78.5	0.9945	0.9806	83.5	0.9858
		4		3.79		74.9	0.9896	0.9782	79	0.9843
		5		5.14		60.9	0.9873	0.9805	63.4	0.9872
		6		5.58		55.2	0.9875	0.9824	57	0.9892
IV	CAF	1	8	1.52	0	88	0.0089	0.9908	99.2	0.9914
		2	8	2.45		102.9	1.0151	0.9876	113.3	0.9906
		3	8	3.52		103.4	1.0111	0.9775	116.6	-
		4	8	4.4		97.8	1.0049	0.9829	106.4	0.9870
		5	16	2.45		88.4	1.0179	0.9952	96.6	0.9984
		6	16	3.52		89.1	1.0122	0.9856	99.1	-
		7	16	4.4		81.9	1.0101	0.9905	89.3	0.9952
		8	24	1.52		62.4	1.0215	1.0058	66.7	1.0091
		9	24	2.45		78.9	1.0153	0.9973	85.3	1.005
		10	24	3.52		77.1	1.0152	0.9944	84.7	0.9984
		11	24	4.4		72.2	1.0063	0.9949	76.4	0.9996

TABLE VI

Effect of the Variation of Fission Integrals on the
Eigenvalue for 13 of the Criticals

$I_{\infty f}^{(239)}$	$\alpha_{\infty}^{(239)}$	k_{eff}	$\nu^{(239)}$	$\eta_{2200}^{(239)}$
338	0.67	1.0005 ± 0.0069	2.869	2.058
308	0.74	1.0000 ± 0.0054	2.886	2.070
293	0.78	1.0008 ± 0.0053	2.898	2.079
278	0.82	1.0013 ± 0.0056	2.909	2.087

The value of $I_{\infty f}^{(239)} = 293$ barn was chosen, not only because it gave the minimum standard deviation in k_{eff} but primarily because it was felt that the corresponding η_{2200} value of 2.079 was more consistent with the value tabulated by Schmidt in KFK 120.

The effect of these normalizations on the calculational values of k_{eff} for each lattice are shown in the last column of Table V. The net effect is that the trend of k_{eff} with lattice pitch has been decreased. No attempt was made to reduce the trend (d) with the content of ^{240}Pu . This is because k_{eff} is not very sensitive to the value of the resonance integral since the epithermal captures in ^{240}Pu are about 2.5% in the experiment CAF-1 and about 5% in the experiment CAF-8. A reduction of the resonance integral by a factor of 2 is necessary to fully eliminate the difference between the corresponding values of k_{eff} . Probably part of the difference is due to the model used for the calculation of leakage. The trend (c) is a very important one for the correlation of the critical experiments. It was previously noticed in the analytical correlation of UO_2 experiments in hexagonal lattices.⁽¹⁾ Probably a different theory for the greyneess correction in the Dancoff factor calculation could fully eliminate the effect. Nevertheless, it should be noticed that H_2O power reactors with two possible exceptions, have a square lattice, and for this kind of lattice the RIBOT model is reasonable accurate.

In summary, using these cross sections the average value for 13 of the critical experiments is $k_{\text{eff}} = 1.0008 \pm 0.0053$. For 13 subcritical measurements the average is $k_{\text{eff}} = 0.9938 \pm 0.0073$.

B. Discussion on the B^2 Measurements for H_2O Lattices

The comparisons shown in Table V indicate that one of the most important quantities required in this type of analysis is an accurate value of the buckling. In fact, all cell codes compute nuclear constants as function of the B^2 term, (i.e., the leakage dependent spectrum) which which is always very small in power reactors. However, in clean lattice critical experiments B^2 is usually larger, as shown in Table V. Thus to compare critical experiments with calculations, a value of B^2 should be supplied. This value can be measured with critical (radial) flux traverses or with subcritical (axial) flux decays, or can be calculated with multigroup diffusion theory or transport theory. From the values shown in Table V, we compare the calculated B^2 with values measured in experiments, I and II (WREC experiments) and exponential experiments, III and IV (PNL experiments). In Fig. 1 the value $\Delta = B_{\text{exp}}^2 - B_{\text{calc}}^2$ is plotted in function of B_{exp}^2 for some of the experiments of these experiments (in units of m^{-2}). The error bars are the experimental errors quoted in the measurements (one standard deviation).

From the results shown in Figure 1, the following conclusions can be drawn:

1. The subcritical B^2 measurements are affected by small statistical errors. They are, instead, affected by a systematic error due to the fact that the "reflector saving" as measured in the exponential

FIGURE 1
Comparison of Techniques used in Measuring Buckling

experiment is higher than the critical reflector saving. A linear fitting deduced from the figure gives $\Delta B^2 = 1.1 + 0.188 (B_{\text{exp}}^2 - 50) \pm 1.3\text{m}^{-2}$.

2. The critical experiments are also affected by a systematic error ($3.2 \pm 2.2\text{m}^{-2}$ on 16 criticals), but this error does not appear to have a trend as a function of the buckling itself. The trouble with critical bucklings is that their measurements is affected by a very big random error, due to the sensitivity of the final value to the single γ measurement of the radial flux.

The next question is: is it possible to measure buckling in such a way to give the theoretician a reliable number to use in his calculations and how to avoid a long, cumbersome and not always reliable criticality calculation? Here, two possible methods are proposed:

1. A completely experimental method: Windsor of BNL (N.S.E., September, 1967) has already proposed it and calls it the variable loading method with variable reflector saving. It consists essentially in measuring exponential shapes as a function of the number of rods in the facility, for example, 50%, 60%, 70%, 80%, and 90% of the critical mass with extrapolation to the full critical mass.

W. P. Stinson (private communication) has already tried this method with 44%, 53% and 60% of the critical mass, although the results cannot be extrapolated to the full critical. His results are in the following table.

TABLE VIIEBWR Rods, Pitch .9" ~ Critical Rods 452 ± 2

<u>Exponential Rods</u>	<u>B^2 (m^{-2})</u>	<u>λ (cm)</u>
192	60.1 ± 0.19	6.99
240	60.7 ± 0.17	6.75
270	61.8 ± 0.18	6.50

Linear extrapolation to 452 rods $B^3 \approx 66.5 \pm 2$ Critical calculation: $B^2 = 63.3 m^{-2}$

In conclusion, the results are too far from criticality to get precise results. A possible handicap of this kind of measurement is that increasing the mass, the correction due to the spontaneous fission becomes more and more important and the experimental error increases.

2. A more precise but cumbersome interpretation of the exponential measurements can be made in many different ways. The simplest one is to calculate with a 1-D diffusion code λ_{radial} as function of the core size using a 1-D diffusion theory code and use this result to correct the measured buckling.

A more reliable way is to simulate the experimental experiments themselves in a 2-D (RZ) calculation, allowing for the presence of the source. The purpose is to reproduce the flux shapes measured in the experiment. Upon matching the flux shapes a calculation is performed for the critical assembly using the same group constants to obtain the critical reflector savings value, $\lambda_{\text{c radial}}$. The difference between the calculated values of λ_{radial} for the exponential and critical assemblies is then used to correct the value measured in the experimental assembly.

The drawbacks to this method are that,

- (1) one must rely on the particular calculation model he uses;
- (2) the interpretation of the experiments become time consuming and expensive.

C. Temperature Coefficients

The calculations of the temperature coefficient measurements for 2% lattices⁽⁹⁾ were made using four groups. The code HFN was used for these calculations of $k_{\text{eff}}(T)$. The calculations were made with a constant reflector temperature (20°C) and a reflector temperature equal to the moderator temperature (as was the case in the measurements). Provision was made in all HFN cases for the thermal expansion of the core (using a thermal expansion coefficient of $9 \times 10^{-5}/^{\circ}\text{C}$ for the lucite templates). The results of the calculation are shown in Fig. 2 and Fig. 3. For both cases the k_{eff} at 20°C was 1.5% lower than the experimental results. Nevertheless, these results (Figs. 2 and 3) show that the temperature coefficients can be reasonably well calculated by RIBOT. The author feels that the good agreement which has been obtained (± 1 pcm/ $^{\circ}\text{C}$) is fortuitous. In fact, as the calculations with the reflector at 20°C show, the measurements of this parameter in a critical facility are too much dependent on the leakage characteristics. In a small core, the reliability of the code to calculate the temperature coefficient, is not expected to be better than several pcm/ $^{\circ}\text{C}$. These calculations of the temperature coefficient can also be interpreted in terms of an increase of the "reflector saving" due to the coupling of the core and the reflector.

The temperature coefficient results can be fitted as follows if the $B^2(T)$ is calculated from $k_{\text{eff}}(T)$ and allowance is made for the increase of the radius.

First calculate the temperature coefficient, C_R with RIBOT for a temperature independent, B^2 . Then correct this value for the expansion of the core radius and the decrease of the buckling with temperature. These are all coupled in the following formula.

$$C = C_R + \frac{(2\alpha R + \delta\lambda_r) B^2 M^2}{R + \lambda_r}$$

C is the fitted value of the temperature coefficient, α is the linear expansion coefficient of the core ($^{\circ}\text{C}^{-1}$), $\delta\lambda_r$ is the reflector savings expansion coefficient. The value of $\delta\lambda_r$ is $2 \times 10^{-3} \text{ cm}/^{\circ}\text{C}$ if the reflector is at a constant 20°C temperature (like in safety calculations). and is $6.4 \times 10^{-3} \text{ cm}/^{\circ}\text{C}$ if the reflector has the same temperature as the core. This formula fits the measured temperature coefficients shown in Figs. 2 and 3 and should be very useful in correlating subsequent measurements.

FIGURE 2 and 3
Comparison at Calculated and Measured Temperature Coefficients for $\text{UO}_2\text{-2wt\% PuO}_2\text{-H}_2\text{O}$ Lattices

IV. INPUT INSTRUCTIONS FOR RIBOT

PROBLEM I.D. AND LIBRARY CARDS

SK 1 Column 1 to 66: Title

Column 67-72: if blank or zero no alterations will be made in the library, if 1, an alteration is made in the library. In this case a new card (SK 1A) follows -

SK 1A (Format 6E 11.5) 6 factors are given which will be multiplied by the following variables:

$\sigma_f^{th}(U^{235})$, $\sigma_a^{th}(U^{235})$, $\sigma_f^{th}(Pu^{239})$, $\sigma_a^{th}(Pu^{239})$ for both thermal groups; v^{235} , v^{239} for all groups.

If a sequence of cases is given, the first four factors are restored to 1 after each case, the last two alteration factors are permanent; i.e., the same two factors will be applied to all the sequence. If, in a successive case, their initial value is to be restored, a new alteration card must be set in a suitable way. Suggested input: do not alter the library.⁽¹⁾

SYSTEM SPECIFICATION CARDS

SK 2 Data cards: K,H(K) (Format 5(I3, E 11.5)) - Data different from zero do not need to be punched - Each datum must be preceded by its corresponding code number - If a succession of cases is given, only data which are different from the preceding case should be given. The same datum could be given many times: only the last one will be used - Every card contains a maximum of 5 data; A K = 80 specified in the last place of some card, ends the sequence of the SK2 cards.

<u>Code #</u>	<u>Datum</u>
1	Fuel pellet radius (cm)
2	Thickness of the gap between fuel and clad (cm)
3	Fuel clad thickness (cm)
4	Fuel rod pitch (cm)
5	Number of fuel rods on the cluster side (#)
6	Box clad thickness (cm)
7	Thickness of the moderator outside the boxes (cm)
8	Cruciform control rod thickness (cm) (square cluster lattice only)
9	Cruciform control rod half span (cm) (square cluster lattice only)
10	Number of cruciform control rods (#) (square cluster lattice only)
11	Number of fuel elements (#)
12	Active height of the fuel (cm)
13	Moderator temperature ($^{\circ}\text{C}$)
14	Fuel effective temperature ($^{\circ}\text{C}$)
15	Coolant (inside boxes) average density (g/cm^3)
16	Moderator (outside boxes) average density (g/cm^3)
17	Area of structural materials external to the cell (cm^2) ⁽²⁾
18	σ_a (2200) of structural materials (barns)
19	Atomic density of structural materials ($\text{atoms}/\text{cm}^3 \times 10^{24}$)
20	Natural boron in solution in the H_2O (parts per million weight of nat. B/weight of H_2O)
21	Time until the first time step H(25) is used (days)
22	Time until the second time step H(26) is used (days)
23	Time until the third time step H(27) is used (days)
24	Time until the fourth time step H(28) is used (days)

<u>Code #</u>	<u>Datum</u>
25	First time step (days)
26	Second time step (days)
27	Third time step (days)
28	Fourth time step (days)
29	Total buckling (cm^{-2})
30 ⁽³⁾	Total weight of the fuel (g): only heavy atoms, excluding oxygen weight
31 ⁽³⁾	Fuel density (g/cm^3) as oxide
33 ⁽⁴⁾	Weight % of ^{233}U with respect to total weight of heavy atoms
34	Weight % of ^{234}U with respect to total weight of heavy atoms
35	Weight % of ^{235}U with respect to total weight of heavy atoms
36	Weight % of ^{236}U with respect to total weight of heavy atoms
39	Weight % of ^{239}Pu with respect to total weight of heavy atoms
40	Weight % of ^{240}Pu with respect to total weight of heavy atoms
41	Weight % of ^{241}Pu with respect to total weight of heavy atoms
42	Weight % of ^{242}Pu with respect to total weight of heavy atoms
43	Weight % of ^{233}Pa with respect to total weight of heavy atoms
50	Fuel cycle option: density of heavy atoms in the recycled fuel (g/cm^3). See SK5B for explanations.
51	% of the old fuel to be recycled.
52	Total weight of the new fuel to be added (g of heavy metals)
54	Number of side rods for Dancoff effect calculation (only for regular square lattice) $H(54) \leq 4$; if both $H(54)$ and $H(55)$ are zero, they are both automatically set to 4 in the code.

<u>Code #</u>	<u>Datum</u>
55	Number of diagonal rods for Dancoff effect calculation (only for regular square lattice) $H(55) \leq 4$; if both $H(54)$ and $H(55)$ are zero, they are both automatically set to 4 in the code.
56	Density of natural boron in the control rods (g/cm^3)
57	Total power (Mw th) or power density (W th/cm^3)
58	If KEVI is 1 or 2, $H(58)$ is the maximum time (days), the end of life k_{eff} , or the end of life k_m for burnup calculations, for $KD4 = 1, 2, \text{ or } 3$ respectively. (See $KD4$)
60	Clad density (g/cm^3)
61 ⁽⁵⁾	Weight % of Al in the clad
62	Weight % of Cr in the clad
63q	Weight % of Mn in the clad
64	Weight % of Ni in the clad
65	Weight % of Zr in the clad
70	Box clad density (g/cm^3)
71 ⁽⁵⁾	Weight % of Al in the box clad
72	Weight % of Cr in the box clad
73	Weight % of Mn in the box clad
74	Weight % of Ni in the box clad
75	Weight % of Zr in the box clad
76	Natural boron density in the fuel as burnable poison (g/cm^3)
77	Natural samarium density in the fuel as burnable poison (g/cm^3)
78	Natural Europium density in the fuel as burnable poison (g/cm^3)

CONTROL CARDS

SK3 Control indexes - Format 6(I3)

KD1 = 1 square regular lattice

 = 2 square cluster lattice

 = 3 hexagonal regular lattice

KD2 = 1 (PuO₂) - UO₂ fuel and burnup equations

 = 2 UO₂ - ThO₂ fuel and burnup equations

KD3 = 1 Diffusion or

 = 2 shock* by shock theory used in the moderator for the

 calculation of the disadvantage factor (Recommended value: 1)

KD4 = 1 life calculation stops when time = H(58)

 = 2 life calculation stops when $k_{\text{eff}} = H(58)$

 = 3 life calculation stops when $k_m = 1/T \int_0^T k_{\text{eff}} dt = H(58)$

 (effective when KEVI = 1 or 2).

KD5 = 0 prints only initial and final results in a burn-up

 calculation

 = N prints also results at time step N, 2N, 3N, ---

KD6 = 1 once through calculation.

 = 2 self-expanding calculations

KD7 (for the package code only; not used in RIBOT)

 = 0 transmits to the economic code (QUICK) only the initial

 and final inventories of materials;

 = N transmits the inventories also at time steps N, 2N, 3N,---

* Meaning first flight collision.

PROBLEM CHANGE CARDS

SK3A For self expanding calculations: the cards must be punched only
 if KD6 = 2 Format (I3/4 (2I3, E11.5))
 NSE: number of self expanding data (if $NSE \leq 4$)
 ISE(K) code of the k-th datum to be self expanded;
 NPSE(K) Number of self expansion steps for the K-th datum;
 PSE(K) Self expansion step in H(K) (in the same units as H(K));
 NSE is given in the first card; ISE, NPSE, PSE in the second card.
 NSE = 5 applies to the special case where a self expansion is made

on plutonium enrichment preserving the original isotopic composition of plutonium as in H(39) - H(42). In this case only one successive set (ISE, NPSE, PSE) must be supplied with ISE = 39, NPSE and PSE corresponding to ^{239}Pu values; the variations in ^{240}Pu , ^{241}Pu , ^{242}Pu will be made accordingly.

NSE = 6 applies when the preceeding option is to be used with two different Pu isotopic compositions. In this case four successive sets (ISE, NPSE, PSE) must be supplied: the first one has the same meaning as in the preceeding option, the other three must contain in ISE the code numbers 40, 41, 42 and in PSE the initial percent of ^{240}Pu , ^{241}Pu , ^{242}Pu to go with H(39) in the second isotopic composition self expansion.

SK4 Control and search indexes. Format (2I3, 3I1.J)

KEVI = 1 Burn-up calculation

= 2 Burn-up calculation with search of the datum which yields
 a life = VIKE

SK4 (continued)

KEVI = 3 k_{eff} calculation

= 4 k_{eff} calculation with search of the datum which yields a

$$k_{\text{eff}} = \text{VIKE}$$

IVAR Code of the datum to be varied;

GUEVAR First variation to be given the datum H(IVAR) at the first attempt;

PREVAR Precision required for the search on VIKE:

VIKE Value, life in days, (KEVI = 2); k_{eff} (KEVI = 4).

If IVAR = 80, the search will be done varying all Pu isotopes while preserving their ratios. In this case the datum GUEVAR refers to ^{239}Pu . The self expanding option and the search option could be used together; of course the self expanding code number must be different from the search code number; for example, a case could be made in which the percent ^{239}Pu is self expanded while searching for the percent in ^{235}U which keeps the burn-up constant.

SK5A Spent fuel card (Format 11 I3); IFV(K): codes of the data for the isotopes of the spent fuel to be recycled (how much of this fuel is to be recycled is specified in H(51)).

SK5B Cards for the fuel make-up (two cards, format 4(I3, E 10.5))
IFA(K) code for the K-th isotope present in the fuel make-up.
($K \leq 7$).

PERCFA(K) weight % of the K-th isotope in the make-up.

If $K \leq 4$, only 1 card SK5B must be supplied.

With this option, a first fuel burn-up will be made; then all fission products and burnable poisons will be separated, and some fuel heavy isotopes (specified in IFV) in a certain percentage (specified in H(51)) will be mixed with a quantity (H(54)) of make-up fuel (specified in SK 5B). The final fuel will have a final density H(50). The balance to reach the final density H(50) will be made up with ^{232}Th or natural uranium (depending on KD 2) and a new fuel cycle will start.

NOTES TO EXPLAIN THE MEANING OF VARIOUS INPUT CARDS

- 1) An alteration of the Library can also be made through the use of the data H(53) and H(59). H(53) changes the value of the fission resonance integral of ^{239}Pu . If no alterations are made ($H(53) = 0$), the value of H(53) is set in RIBOT to 164.8. An alteration is made by giving a different value in input. For example: if $I_{f_{\infty}}(239)$ has to be 303 b (10 b more than the standard value), then $H(53) = 174.8$. H(59) is a multiplier factor for ^{240}Pu resonance integral. If no input value is specified, a value of 1 is automatically set in the code. The alterations are temporary, i.e., in the new cases the standard values are restored.
- 2) For square cluster lattices only, structural materials (shrouds, control curtains or similar) are allowed to take place in the moderator outside the boxes. The area, atom concentration, and average microscopic cross sections are given. A "core" spectrum is used for the calculation of the absorption cross section and the external moderator flux is used for homogenization purposes. The structure nuclei do not have

an epithermal resonance cross section in excess of $1/v$ cross section. If the structural materials are self-shielded with respect to the moderator, allowance should be made in the specified concentrations or sections. The structure area is subtracted from the area of the external moderator. No burnup calculations are made for structural nuclei.

- 3) The data H(30) and H(31) exclude, of course, each other.

If H(31) is zero, then one must specify:

H(30): total fuel weight;

H(11): total number of fuel clusters;

H(12): active weight of the fuel;

H(57): total thermal power;

in this case, if H(29) (buckling) is zero, a reflector saving is calculated with an empirical formula.

If H(31) is given > 0 , then H(30), H(11), H(12) could be zero and H(57) is the power density; in this case, H(29) is not calculated, and must be specified in input.

The first option is generally used for power reactors, while the second one is used in lattice experiment calculations.

- 4) The balance to 100% in the fuel is made up of ^{238}U or ^{232}Th for KD2 = 1 or 2 respectively.

- 5) The balance to 100% is made up of Fe.

V. OUTPUT DESCRIPTION

The output is self-explanatory. The following is valid for a few somewhat ambiguous captions.

In the first page of the output only input numbers are printed; in the second (and sometimes the third) page, the volumes and weights are printed: Here T is a metric ton (1000 kg); in the successive page(s) the calculated nuclear constants are printed at the specified times.

The atom densities are meant in the material (not averaged in the core). The fast microscopic cross sections have been already collapsed to one fast group. UNIV is the cross section of a nucleus having a 1 barn resonance integral.

In the fast group, the microscopic cross sections U235A, Pu239A, etc. are the fast capture cross sections and do not include the fission cross sections.

UNIT is the thermal cross section of a nucleus with a 1/v behavior and a value of 1 barn at 2200 m/sec.

In the thermal groups the absorption term includes the fission cross section.

In both five and two group constants, SIGMA-A includes (if present) the poison cross sections of the control materials (boron in solution or control rods). SIGMA-P includes the absorption of these materials and does not include burnable poison cross sections.

P is the resonance escape probability for the principal fertile nuclide (^{238}U or ^{232}Th). P-TOT is the absorption escape probability for all fast absorptions.

MWD/TON is the energy produced by the reactor in thermal Megawatt-days per metric ton of initial heavy metal. INIT-C is the instantaneous conversion ratio, defined as fissile atoms produced/fissile atoms destroyed. Of course, this number is initially zero in a Th-U fuel cycle because of

the 30 day lag in the production of ^{233}U due to the decay of ^{233}Pa . The corresponding two day delay in U-Pu fuel cycle due of ^{239}Np was not allowed for.

INTEGR.C. is one of the possible definitions of the integral conversion ratio. This particular one is defined as

$$\text{INTEGR.C.} = 1 - \frac{\text{Initial fissile atoms} - \text{Final fissile atoms}}{\text{Burned fissile atoms}}.$$

HOM.FACT.4 and HOM.FACT.5 are calculated as $V_{\text{fuel}} + V_{\text{clad}} F_{\text{clad}}^{4,5} + V_{\text{cool.}} F_{\text{cool.}}^{4,5} + V_{\text{box}} F_{\text{box}}^{4,5} + V_{\text{ext.mod}} F_{\text{ext.mod.}}^{4,5}$, where F^4 and F^5 are the disadvantage factors for thermal groups 4 and 5 and V is the volume fraction. These homogenization factors are handy to convert materials cross sections to reactor average macroscopic cross sections.

2200 SECT.FP is the macroscopic 2200 m/sec absorption cross section of the pseudo fission products in the fuel (i.e., not yet homogenized in the reactor).

SAT.FP.GR4 and SAT.FP.GR5 are the macroscopic cross sections of the high cross section fission products (^{135}Xe , ^{113}Cd , ^{149}Sm , ^{151}Sm , ^{155}Eu , ^{157}Gd) averaged over the two thermal groups (in the fuel).

PEAK FACT is the local peak factor in the fuel cluster due to the water gaps. It is a fair estimate if the fuel is uniformly distributed acrosss the cluster.

A warning message is printed in case of alterations of the library.

The control indexes are checked. The case is ended if they are out of the permissible range.

The calculation time is about 1 sec of UNIVAC 1108 for the first time step and about .2 sec for each subsequent time step. For IBM 7094 these times are 5 times bigger.

VI. ACKNOWLEDGMENTS

As was mentioned in the foreword, the construction of the physical model and the mathematical formulation of the problem was done by the author in cooperation with G. Buffoni and S. Lopez who appear as co-authors of the original paper concerning the RIBOT code. The author wishes also to thank Veikko Uotinen for the helpful discussions on the temperature coefficients measurements and calculations. Finally, the author acknowledges the valuable work done by W. Len Purcell for the transformation of the initial IBM 7094 code to the actual UNIVAC 1108 code and for the adjustment of some of the complicated options built in RIBOT.

VII. REFERENCES

1. G. Buffoni, P. Loizzo, S. Lopez, M. Petilli. Burn-up of Pressurized and Boiling Water Reactors, Symposium on Advances of Reactor Theory, Karlsruhe, June 1966. CNEN Report RT/FI(67)28 (1967).
2. R. M. Hiatt, "REPP - A Thermal Hydraulics Design Code for Light Water Power Reactors," (Unpublished Data)
3. E. A. Eschbach, et al. "QUICK - A Simplified Fuel Cost Code," HW-71812, January (1962).
4. G. Forti, Nuc. Sci. and Eng., 19, 449 (1964).
5. E. Cupini, A. DeMatteis, P. Loizzo, Energia Nucleare, 14, 321 (1967)
6. R. D. Leamer et al. $\text{PuO}_2\text{-UO}_2$ Fueled Critical Experiments, WACP-2726-1 (1967).
7. E. G. Taylor, et al. Saxton Plutonium Program - Critical Experiments for the Saxton Partial Plutonium Core, WCAP-3385-54 (1965).
8. L. C. Schmid, et al. Critical Masses and Bucklings of $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Systems, HW_83187 (1964).
9. W. P. Stinson. Private Communication.
10. V. O. Uotinen, L. D. Williams. Experiments and Calculations for H_2O -Moderated Assemblies Containing $\text{UO}_2\text{-2 wt\% PuO}_2$ Fuel Rods. BNWL-SA-1107 (1967).
11. H. C. Honeck, "ENDF/B, Specifications for Evaluated Nuclear Data File for Reactor Applications," BNL-50066(T-467), ENDF-102, May 1966, Revised July 1967.
12. G. Buffoni, S. Lopez, I. Maganzani, "BURNY: A 2-Dimensions Burn-up Code" (in Italian) - CNEN RT/FI(67)8 (1967).

APPENDIX A

SAMPLE CASE

INPUT CARDS: THE FIRST SEVEN CARDS AND THE LAST CARD ARE NECESSARY FOR THE OPERATION OF THE UNIVAC 1108 COMPUTER; IN PARTICULAR, THE SIXTH CARD CALLS THE LIBRARY.

```

*   RUN  5B      ,CUS235,3,1000          P. LOIZZO
*RX  ASG  B=U9345
*   XQT  CUR
IN B
TRI B
*   DAT  RIBOT5/LOIZZO,0
*   XQT  RIBOT
      RIBOT-5 TEST CASE FOR BNWL REPORT
      1 .6355          3 .081          4 1.8745          5 7.          6 .1
      7 .959          11 764.          12 366.          13 286.          14 600.
      15 .48          16 .77          21 3000.          25 60.          30 1.50869+08
      35 2.57          57 3293.          58 1.0586          60 6.5          65 100.
      70 6.5          75 100.          25 60.          80          80
      2 1 1 3 0 1
      1
*   FIN

```

RIGBT-5 TEST CASE FOR BNWL REPORT

GEOMETRICAL CHARACTERISTICS-LENGTHS IN CM

SQUARE CLUSTER PITCH	7.000*	7.000		
FUEL RADIUS	.6355		GAP THICK.	.0000
CLAD THICK.	.0810		FUEL ROD PITCH	1.8745
BOX THICK.	.1000		EXT. MOD. THICK.	.9590
FUEL EL. NUMB.	764.		ACTIVE HEIGHT	366.00

MATERIALS CHARACTERISTICS

FUEL UO2

PERCENT COMPOSITION

U 235 = 2.57000

U 238 = 97.43000

FUEL CLAD MADE OF ZIRC

DENSITY = 6.50 GR/CM3

FUEL BOX MADE OF ZIRC

DENSITY = 6.50 GR/CM3

HERMAL-HYDRAULIC CHARACT.

THERMAL POWER 3493.000 MW

FUEL TEMPERATURE 600. C

COOLANT DENSITY .480 GR/CM3

COOLANT TEMP. 286. C

EXT. MOD DENSITY .770 GR/CM3

RIBAT-5 TEST CASE FOR BNL REPORT

VOLUMES IN THE ELEMENTARY CELL (CM**3)

FUEL	GAP	CLAD	TOT. VOL.
464.368	.000	125.919	590.287

VOLUMES IN THE FUEL ELEMENT (LITERS)

FUEL	GAP	CLAD	COOLANT	BOX	TOT. VOL.
22.754	.000	6.170	34.092	1.936	64.951

VOLUMES IN CORE (M**3)

FUEL	GAP	CLAD	COOLANT	EXT. MOD.	TOT. MOD.	BOX	VOL. CORE
17.3841	.0000	4.7139	26.0459	15.3178	41.3637	1.4788	64.9405

VOLUME RATIOS IN THE ELEMENTARY CELL

FUEL	GAP	CLAD	COOLANT
.3611	.0000	.0979	.5410

VOLUME RATIOS IN THE CORE

FUEL	GAP	CLAD	COOLANT	EXT. MOD.	BOX
.2677	.0000	.0726	.4011	.2359	.0228

ATOM DENSITIES IN AT./CMBARN

	AL	CR	FE	MN	NI	ZR
CLAD	0.0000	0.0000	0.0000	0.0000	0.0000	4.2968-02
BOX	0.0000	0.0000	0.0000	0.0000	0.0000	4.2968-02
	H	O				
COOLANT	5.4160-02	1.6080-02				
EXT. MOD.	5.4590-02	2.1795-02				

WEIGHTS

CLAD 30.040 T

MODERATOR 24.297 T

BOX 9.612 T

A-4

BNWL-735

RIBOT-5 TEST CASE FOR BNWL REPORT

TIME 0. DAYS

ATOM DENSITIES IN ATOMS/CM³ AND WEIGHTS IN KG (FUEL MATERIALS)

U235	U238	U238	PU239	PU240	PU241	PU242	0
5.7165-04	0.0000	1.1401-02	0.0000	0.0000	0.0000	0.0000	4.3945-02
5.8773+03	0.0000	1.4699+05	0.0000	0.0000	0.0000	0.0000	0.0293+04

FAST MICROSCOPIC CROSS SECT. IN BARN

UNIT	U235A	U235F	U238A	U238F	PU239A	PU239F	PU240A	PU241A	PU241F	PU242A
.03157	4.33651	8.49493	10.97946	.73730	7.20446	9.26303	263.80225	5.75288	18.17478	62.55387

THERMAL MICROSCOPIC CROSS SECT. GROUPS 4-5 IN BARN

UNIT	U235A	U235F	U238A	U238F	PU239A	PU239F	PU240A	PU241A	PU241F	PU242A
2.85723-01	1.80955+02	1.49101+02	2.28089+03	1.35187+03	1.69223+02	8.63652+02	1.08306+01	2.41716+01	H TR	H SC
6.47681-01	4.10541+02	3.48555+02	8.64655+02	5.85989+02	1.97597+02	7.63870+02	2.59873+01	3.73736+01	H TR	H SC

FIVE GROUP CONSTANTS

GROUP	D	SIGMA-A	SIGMA-R	SIGMA-F	NU-SIGMA-F	SIGMA-P
1	2.32196+00	1.73248-03	4.39646-02	1.29136-03	3.43732-03	0.00000
2	1.13409+00	0.00000	1.15912-01	0.00000	0.00000	0.00000
3	9.81389-01	2.27323-02	5.85290-02	4.53123-03	1.11015-02	0.00000
4	6.75316-01	3.18610-02	3.26086-01	2.01161-02	4.92844-02	0.00000
5	3.36709-01	6.53414-02	4.12196-02	4.00004-02	9.80011-02	0.00000

TWO GROUP CONSTANTS

GROUP	D	SIGMA-A	SIGMA-R	SIGMA-F	NU-SIGMA-F	SIGMA-P
1	1.69845+00	7.40886-03	1.67912-02	1.96136-03	4.94474-03	0.00000
2	4.05711-01	5.89216-02	0.00000	3.59512-02	8.80804-02	0.00000

K-EFF	K-INF	EPSILON	ETA*F	P	P-TOT	B**2	TAU	L**2	M**2
1.22591	1.24154	1.04947	1.49437	.81542	.69385	1.68	70.18	6.99	77.06

WD/TOT	INIT.C.	INTEGP.C.	HOM. FACT.4	HOM. FACT.5	2200 SECT.FP	SAT.FP. GR4	SAT.FP. GR5
0.0000	0.0000	0.0000	1.1342+00	1.3334+00	0.0000	1.5133-03	1.6332-02

DELTA-25	DELTA-26	RO-26
7.5865-02	1.6126-02	1.3029+00

DISADVANTAGE FACT. FOR GROUPS 4 AND 5

CLAD	COEFFICIENT	ROX	EXT.MOD	INT.ROX	PEAKFACT
1.05517	1.09030	1.28611	1.36074	1.05750	1.27336
1.11390	1.20494	1.67217	1.06514	1.12203	1.51802

A-5

BNWL-735

PILOT-5 TEST CASE FOR BNWL REPORT

TIME 1467. DAYS

ATOM DENSITIES IN ATOMS/CM BARN AND WEIGHTS IN KG (FUEL MATERIALS)

U235	U236	U238	PU239	PU240	PU241	PU242	O
1.0267-04	7.5227-05	2.0983-02	8.4367-05	5.0289-05	2.0641-05	9.0015-06	4.3945-02
5.9639+02	5.1182+02	1.4412+05	5.8085+02	3.4795+02	1.4341+02	6.2801+01	2.0293+04

FAST MICROSCOPIC CROSS SECT. IN BARNs

UNIV	U235A	U235F	U236A	U238A	PU239A	PU239F	PU240A	PU241A	PU241F	PU242A
.03020	4.32839	8.35471	8.46118	.70638	6.70953	8.69629	83.74825	5.50295	17.37387	56.52614

THERMAL MICROSCOPIC CROSS SECT. GROUPS 4-5 IN BARNs

UNIT	U235A	U235F	PU239A	PU239F	PU240A	PU241F	H TR	H SC
2.85598-01	1.80939+02	1.49106+02	2.22994+03	1.32202+03	1.69332+02	8.67544+02	1.08709+01	2.42088+01
UNIT	U235A	U235F	PU239A	PU239F	PU240A	PU241F	H TR	H SC
6.54497-01	4.15576+02	3.52958+02	8.65247+02	5.87502+02	1.99006+02	7.66370+02	2.62626+01	3.76023+01

FIVE GROUP CONSTANTS

GROUP	D	SIGMA-A	SIGMA-R	SIGMA-F	NU-SIGMA-F	SIGMA-P
1	2.32126+00	1.69914-03	4.39646-02	1.26653-03	3.37019-03	0.00000
2	1.13409+00	0.00000	1.15912-01	0.00000	0.00000	0.00000
3	9.31389-01	2.76582-02	5.85290-02	1.90226-03	5.19455-03	0.00000
4	6.28952-01	5.98552-02	3.44723-01	3.09433-02	8.88276-02	0.00000
5	3.45139-01	5.30358-02	3.93817-02	2.16489-02	5.98260-02	0.00000

TWO GROUP CONSTANTS

GROUP	D	SIGMA-A	SIGMA-R	SIGMA-F	NU-SIGMA-F	SIGMA-P
1	1.71102+00	8.47551-03	1.60617-02	1.18211-03	3.18183-03	0.00000
2	3.93896-01	5.42073-02	0.00000	2.32456-02	6.48083-02	0.00000

K-EFF	K-INF	EPSILON	ETA*F	P	P-TOT	B**2	TAU	L**2	M**2
.90079	.91227	1.04845	1.19556	.81867	.65459	1.68	69.73	7.27	77.00

AWD/TON	INIT.C.	INTEGR.C.	HOM. FACT.4	HOM. FACT.5	2200 SECT.FP	SAT.FP. GR4	SAT.FP. GR5
2.7655+04	7.3773-01	4.3381-01	1.2523+00	1.2566+00	2.5246-02	1.1945-03	1.1058-02

DELTA-25	DELTA-28	RO-28
7.3520-02	9.2715-02	1.2333+00

DISADVANTAGE FACT. FOR GROUPS 4 AND 5

CLAD	COOLANT	BOX	EXT.MOD	INT.BOX	PEAKFACT
1.10545	1.19451	1.48878	1.65905	1.11556	1.37630
1.08700	1.15420	1.53058	1.74761	1.09194	1.43805

A-6

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