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SPIC-1 — AN IBM-704 CODE TO CALCULATE THE NEUTRON DISTRIBUTION OUTSIDE A RIGHT-CIRCULAR CYLINDRICAL SOURCE

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P. Gillis

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The SPIC-1 code calculates the fast-neutron dose rate or the thermal neutron flux at a point outside a right circular cylindrical source which is surrounded by cylindrical shell shields and is capped by plane slab shields. The fast neutron attenuation kernel is empirical and is in the form of a linear combination of single exponentials which has been fitted to the experimental fast-neutron dose rate distribution in pure water. Empirical neutron removal cross sections are used to represent the attenuation by shells of non-hydrogenous materials located in the water. Typical computing and editing time for a 20-field-point-problem, in which there are 10 side and 10 top shields, is 6.5 minutes. The code requires an IBM-704 EDPM with a memory of 32,768 words.

SPIC-1—AN IBM-704 CODE TO CALCULATE THE NEUTRON DISTRIBUTION OUTSIDE A RIGHT-CIRCULAR CYLINDRICAL SOURCE

P. Gillis

I. INTRODUCTION

The SPIC-1 code is designed to calculate the fast-neutron dose rate or thermal-neutron flux in water outside a core in the shape of a right circular cylinder. The cylinder contains a source of fission neutrons which varies in the radial and axial directions only. Field points may be located outside the core, in a plane through the axis of the cylinder. The intended application of the code is to the design of power reactor shields which are predominantly hydrogenous. The method of calculation and preparation of input is similar to that of SPAN-2 (see Ref 1).

II. THE CODE FORMULATION

The code finds the fast-neutron dose rate or the thermal-neutron flux at a field point outside a right-circular cylinder of finite height which contains a source distribution S(r, z) in up to three concentric cylindrical regions. The source cylinder is surrounded by cylindrical shell shields of infinite height and is capped by plane slab shields. It is assumed that:

1) The field points, at which the dose rate or flux is to be computed, are located within or beyond hydrogenous regions, so that the use of effective removal cross sections is valid.
2) The source distribution is independent of the azimuthal angle. The units for the source are fission neutron/cubic centimeter-second. Note that the units of distance must be centimeters and the units of time must be seconds.
3) All field points must be in the plane $\theta = 0$.

The origin of the cylindrical coordinate system is placed at the center of the core and the following notation is used:

R_c	Radius of the core
R_n	Outer radius of the n^{th} cylindrical shield region

H	Height of the core
$h_0 = H/2$	Half-height of core (z coordinate of all points on the upper face of the core)
$P_f(r_f, 0, z_f)$	Field point
h_m	Height (from the center of the core) of the upper face of the m^{th} slab shield region
$P(r, 0, Z)$	Point in core
N_S	Number of cylindrical shield regions passed through in moving from P to P_f
N_t	Number of slab shield regions passed through in moving from P to P_f
D	Distance from P to P_f
D_n^S	Distance traveled through the n^{th} cylindrical shield region in moving from P to P_f
D_m^t	Distance traveled through the m^{th} slab shield region in moving from P to P_f
μ_n^S	Factor used to indicate the attenuation of neutrons in the n^{th} cylindrical shield region
μ_m^t	Factor used to indicate the attenuation of neutrons in the m^{th} slab shield region

If exponential attenuation is assumed, the uncollided flux at the field point $P_f(r_f, 0, z_f)$ is

$$\varphi = \int_V \frac{S(r,z)}{4\pi D^2} \exp\left\{-\left[\sum_{n=1}^{n=N_S} \mu_n^S D_n^S + \sum_{m=1}^{m=N_t} \mu_m^t D_m^t\right]\right\} dV. \tag{1}$$

The assumption of exponential attenuation in water is not valid for fast-neutron dose rate or thermal-neutron flux. However, these quantities can be approximated by a fit of the form

$$a_1\varphi_1 + a_2\varphi_2 + a_3\varphi_3 \quad ,$$

where φ_1, φ_2, and φ_3 have the same form as φ in Eq (1), and a_1, a_2, and a_3 are empirically determined constants.

Thus, for both the fast-neutron dose rate and thermal-neutron flux, the code forms the integral

$$\int_V \frac{S(r,z)}{4\pi D^2}\left[a_1 f_1(r,\theta,z) + a_2 f_2(r,\theta,z) + a_3 f_3(r,\theta,z)\right] dV \ , \tag{2}$$

where

$$f_i(r,\theta,z) = \exp\left\{-\left[\sum_{n=1}^{n=N_S} \mu_n^{S_i} D_n^S + \sum_{m=1}^{m=N_t} \mu_m^{t_i} D_m^t\right]\right\}$$

The numbers $\mu_n^{S_i}$ and $\mu_m^{t_i}$ are calculated for each exponential by the formula

$$\mu_n^{S_i} = \sum_{j=1}^{j=N_{S_n}} d_{n,j}^S \lambda_{n,j}^i \ , \tag{3}$$

where N_{S_n} is the number of elements in the n^{th} cylindrical shield region, $d_{n,j}^S$ is the number density of the j^{th} element in the n^{th} cylindrical shield region, and $\lambda_{n,j}^i$ is equivalent to a microscopic cross section for the j^{th} element in the n^{th} cylindrical shield region. (The λ's are contained in the library; the d's are part of input.)

Three sets of numbers may be used to calculate the μ's. The first is for the fast-neutron dose rate in water, the second for the thermal-neutron flux in low temperature water, and the third is for the thermal-neutron flux in high temperature water. For all elements except water (and consequently hydrogen), the λ's are effective removal cross sections, and their use in this case depends upon the assumption that all regions containing non-hydrogenous elements are backed by a sufficient amount of water. The values of the λ's for water and hydrogen are determined empirically and are related to a_1, a_2, and a_3 in each case (see Ref 2).*

The integrals are approximated numerically by Gaussian quadrature. For details of the integration and the calculation of distances, see Ref 1.

The units for the output (which is labeled flux) are millirem/hour for fast-neutron dose rate and neutrons/cm^2-sec for thermal-neutron (2200 m/sec) flux.

III. PREPARATION OF INPUT

The input for SPIC-1 is the same as that of SPAN-2 with the following exceptions:

1) Columns 66 and 67 of the problem title card should contain the digits 01, 02, or 03, depending on whether the fast-neutron dose rate in water, the thermal neutron flux in low temperature water, or the thermal neutron flux in high temperature water is desired.
2) Columns 68, 69, 70, 71, and 72 should contain the characters SPIC-1.
3) On card 1000, word 2 of the SPAN input should not appear, and therefore the SPAN words 3, 4, 5, 6, and 7 become words 2, 3, 4, 5, and 6, respectively, in the SPIC deck.
4) Card 2002 does not appear.

The source deck is prepared in exactly the same way in SPIC-1 as in SPAN-2. In the thermal-neutron flux calculation, low temperature water is water at 20°C, and high temperature water is water at temperatures between 230°C and 290°C.

IV. OPERATING INSTRUCTIONS

The operating instructions are the same in SPIC-1 as in SPAN-2.

V. CODE LIMITS

The limitations of SPIC-1 are the same as those of SPAN-2.

APPENDIX I: THE LIBRARIES

There are three libraries attached to SPIC-1; one must be selected by placing 01, 02, or 03 in columns 66 and 67 of the problem title card. The kernel for library 02 (thermal-neutron flux in low temperature water) was derived for water at 20°C. The kernel for library 03 (thermal-neutron flux in high temperature water) was derived for water at 260°C and has been determined to be valid for water at temperatures from 230°C to 290°C.

The attenuation factor for each region and each exponential is calculated by the use of material numbers (from card series 4000 and 4500) and the number densities (from card series 6000 and 6500). The units of the numbers in the libraries are barns. Thus, the units of the number densities must be 10^{-24} x (atoms or molecules/cm^3). The resultant units of the attenuation factors are 1/cm. An exception is concrete. For concrete, the units of the numbers in the libraries are cm^2/g and the input densities for concrete must be mass density (i.e., g/cm^3).

* Also from unpublished work by K. Shure of Bettis Laboratory on fast neutron penetration (1958).

The materials contained in the library and their associated library numbers are:

1. Hydrogen
2. Oxygen
3. Zirconium
4. Carbon
6. Iron
7. Nickel
9. Aluminum
11. Chromium
18. Uranium
48. Lead
75. Ordinary concrete
77. Water

The data for water and hydrogen have been obtained from Ref 2.* The data for ordinary concrete have been obtained from Ref 3. The data for all other materials have been obtained or inferred from Ref 4. These data are listed in Table I.

TABLE I

TABLE OF LIBRARIES

Library Number and Element	Library No. 01 (Fast Neutrons)			Library No. 02 (Low Temp Thermal Neutrons).			Library No. 03 (High Temp Thermal Neutrons)		
	1	2	3	1	2	3	1	2	3
1 Hydrogen	0.97	1.84	3.57	1.867	0.9611	0	1.175	2.355	3.678
2 Oxygen	0.99	0.99	0.99	0.99	0.99	0.99	0.99	0.99	0.99
3 Zirconium	2.32	2.32	2.32	2.32	2.32	2.32	2.32	2.32	2.32
4 Carbon	0.81	0.81	0.81	0.81	0.81	0.81	0.81	0.81	0.81
6 Iron	1.98	1.98	1.98	1.98	1.98	1.98	1.98	1.98	1.98
7 Nickel	1.89	1.89	1.89	1.89	1.89	1.89	1.89	1.89	1.89
9 Aluminum	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31
11 Chromium	1.83	1.83	1.83	1.83	1.83	1.83	1.83	1.83	1.83
18 Uranium	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6
48 Lead	3.53	3.53	3.53	3.53	3.53	3.53	3.53	3.53	3.53
75 Concrete	0.0397	0.0397	0.0397	0.0397	0.0397	0.0397	0.0397	0.0397	0.0397
77 Water	2.934	4.79	8.47	4.662	2.847	0	3.338	5.699	8.344
Multipliers	0.0316	0.221	-0.1275	17.2	0.357	0	2.83	59.5	-117.0

* Also from unpublished work by K. Shure of Bettis Laboratory on fast neutron penetration (1958).

```
TEST PROBLEM FOR SPIC--  21 SIDE SHIELDS AND 5 TOP SHIELDS       01SPIC1
PRODUCT SOURCE   22 BY  50 MESH
+100+22+50+0000000000000000000000000000000000000000000000000000000000
+200+51219+51219+51219+51219+51219+51219+0000000000000000000000000000
+201+51219+51219+51219+51219+51219+51219+0000000000000000000000000000
+202+51219+51219+51215+51215+51215+51215+0000000000000000000000000000
+203+51215+51215+51215+51127+51215+0000000000000000000000000000000000
+350+50635+0000000000000000000000000000000000000000000000000000000000
+400+509302+509303+509313+509319+509324+509336+0000000000000000000000
+401+509347+509353+509380+509397+509430+509492+0000000000000000000000
+402+509643+511009+511009+511045+511034+511001+0000000000000000000000
+403+509553+509151+511084+511475+000000000000000000000000000000000000
+500+502980+502733+502539+502378+502230+504097+0000000000000000000000
+501+503983+503946+503966+504028+504120+504234+0000000000000000000000
+502+504365+504503+504661+504820+504985+505155+0000000000000000000000
+503+505327+505503+505681+505861+506043+506226+0000000000000000000000
+504+506412+506599+506788+506978+507170+507365+0000000000000000000000
+505+507561+507760+507962-508167+508376+508589+0000000000000000000000
+506+508799+509008+509222-509431+509637+509840+0000000000000000000000
+507+511004+511024+511045+511066+511087+511110+0000000000000000000000
+508+511130+511147+511169+511188+511205+511221+0000000000000000000000
+1000+21+5+50635+50635+51215+0000000000000000000000000000000000000000
+2000+000+000+000+000+000-000+000000000000000000000000000000000000000
+2001+1+524598+000000000000000000000000000000000000000000000000000000
+3000+2+1+1+1+1+1+1+1+1+1+1+1+0+1+0+1+1+1+2+0+0+0+0000000000000000000
+4000+3+77+77+77+6+77+6+77+6+77+6+77+6+6+6+6+77+6+77+11+000000000000000
+4500+77+77+6+77+6+77+00000000000000000000000000000000000000000000000
+5000+521335+51159+51476+51238+51254+51381+00000000000000000000000000
+5001+51254+518+51254+5032+521334+51476+00000000000000000000000000000
+5002+508+51922+51159+524445+51349+528065+000000000000000000000000000
+5003+51349+521524+51127+00000000000000000000000000000000000000000000
+5500+521125+5163+51254+50787+522166+00000000000000000000000000000000
+6000+492465+491132+492715+492715+49848+492715+0000000000000000000000
+6001+49848+492715+49848+492715+49848+492715+000000000000000000000000
+6002+49848+49848+49848+49848+49334+49848+000000000000000000000000000
+6003+49358+49675+000000000000000000000000000000000000000000000000000
+6500+492715+492715+49848+492715+49848+492715+00000000000000000000000
+7000+5326721+5259436+000+000+000000000000000000000000000000000000000
```

TEST PROBLEM FOR SPIC-- 21 SIDE SHIELDS AND 5 TOP SHIELDS   01SPIC1   1 PAGE 1
PRODUCT SOURCE 22 BY 50 MESH

CORE RADIUS   CORE HEIGHT

46.9830   31.1150

LIMITS OF INTEGRATION.	R1	R2	Z1	Z2	THETA1	THETA2.
	.0000	46.9830	-15.5575	15.5575	.0000	3.1416

TABLE OF INTERPOLATED SOURCE.

Z \ R	7479	1 3852	1 9083	2 1587	2 2349	2 3111	2 3790	2 4313	2 4623
2 1527	2668	2670	2673	2682	2702	2894	2931	2648	3570
2 1407	2309	2311	2314	2321	2339	2505	2537.	2292	3090
2 1198	3743	3745	3750	3763	3791	4061	4112	3715	5009
1 9137	3844	3846	3858	3864	3893	4170	4223	3815	5144
1 5723	4558	4561	4567	4582	4617	4945	5008	4524	6100
1 1950	5525	5527	5535	5553	5595	5993	6069	5483	7393
1-1950	6593	6596	6605	6627	6677	.7151	7243	6543	8822
1-5723	7696	7700	7710	7736	7795	8348	8455	7638	1 1030
1-9137	8751	8755	8763	8796	8863	9493	9614	8685	1 1171
2-1198	9596	9601	9613	9645	9719	1 1041	1 1054	9524	1 1284
2-1407	1 1025	1 1026	1 1028	1 1030	1 1038	1 1112	1 1126	1 1017	1 1372
2-1527	1 1060	1 1060	1 1063	1 1065	1 1073	1 1150	1 1164	1 1052	1 1418

SIDE REGIONS

REGION	THICKNESS	OUTER RADIUS	ATTENUATION FACTORS			MATERIAL	NUMBER DENSITY
	46.9800	46.9800	.0904	.1114	.1531	3	.0247
						77	.0113
1	2.1500	49.1300	.0797	.1300	.2300	77	.0271
2	13.3500	62.4800	.0797	.1300	.2300	77	.0271
3	1.5900	64.0700	.1679	.1679	.1679	6	.0848
4	4.7600	68.8300	.0797	.1300	.2300	77	.0271
5	2.3800	71.2100	.1679	.1679	.1679	6	.0848
6	2.5400	73.7500	.0797	.1300	.2300	77	.0271
7	3.8100	77.5600	.1679	.1679	.1679	6	.0848
8	2.5400	80.1000	.0797	.1300	.2300	77	.0271
9	8.0000	88.1000	.1679	.1679	.1679	6	.0848
10	2.5400	90.6400	.0797	.1300	.2300	77	.0271
11	.3200	90.9600	.1679	.1679	.1679	6	.0848
12	13.3400	104.3000	.1679	.1679	.1679	6	.0848
13	4.7600	109.0600	.0000	.0000	.0000		
14	.8000	109.8600	.1679	.1679	.1679	6	.0848
15	9.2200	119.0800	.0000	.0000	.0000		
16	1.5900	120.6700	.1679	.1679	.1679	6	.0848
17	44.4500	165.1200	.0980	.1600	.2829	77	.0334
18	3.4900	168.6100	.1679	.1679	.1679	6	.0848
19	80.6500	249.2600	.2286	.2950	.4268	77	.0358
						12	.0675
20	3.4900	252.7500	.0000	.0000	.0000		
21	15.2400	267.9900	.0000	.0000	.0000		
22	1.2700	269.2600	.0000	.0000	.0000		

TOP REGIONS

REGION	THICKNESS	UPPER HEIGHT	ATTENUATION FACTORS			MATERIAL	NUMBER DENSITY
1	.6350	15.1925	.0797	.1300	.2300	77	.0271
2	11.2500	27.4425	.0797	.1300	.2300	77	.0271
3	6.3000	33.7425	.1679	.1679	.1679	6	.0848
4	2.5400	36.2825	.0797	.1300	.2300	77	.0271
5	.7870	37.0695	.1679	.1679	.1679	6	.0848
6	21.6600	58.7295	.0797	.1300	.2300	77	.0271

.TEST PROBLEM FOR SPIC-- 21 SIDE SHIELDS AND 5 TOP SHIELDS 01SPIC. 1 PAGE 3

..FLUX AT FIELD POINT..

	R	Z.
-17 5456	267.2100	59.4360

..ALL FIELD POINTS DONE,PROBLEM COMPLETED..

END OF RUN,PROBLEM COMPLETED.PRINT OUTPUT TAPE 5 AND IF REQUESTOR DESIRES,SAVE RESTART TAPE 2.SAVE TAPE 7,IF ANY..

REFERENCES

1. P. A. Gillis, T. J. Lawton, and K. W. Brand; "SPAN-2—An IBM-704 Code to Calculate Uncollided Flux outside a Circular Cylinder," Bettis Atomic Power Laboratory Report, WAPD-TM-176 (August 1959).

2. D. C. Anderson and K. Shure; "Calculation of Thermal Neutron Fluxes in Primary Shields," Bettis Atomic Power Laboratory Report, WAPD-TM-193, to be published.

3. U. S. National Bureau of Standards Handbook 63, "Protection Against Neutron Radiation up to 30 Million Electron Volts," (November 22, 1955).

4. G. T. Chapman and C. L. Storrs, "Effective Neutron Removal Cross Sections for Shielding," AECD-3978 (September 19, 1955).