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REACTOR PHYSICS CONSTANTS FOR THE MEDIUM ENRICHED-URANIUM CORE OF KYOTO UNIVERSITY CRITICAL ASSEMBLY

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Abstracts

Two energy-group eleven-region constants are prepared for the single cylindrical core of Kyoto University Critical Assembly (KUCA) loaded with uranium fuel of fourty-five per cent enrichment. The procedure is based on the twenty-six-group microscopic cross section libraries MGCL26GR and SMF26GR as its starting point, and on the flux calculations by one-dimensional transport theory code ANISN-JR. Cross sections are averaged with these flux as weighting functions.

Although the constants do not produce a complete agreement with experiment when used for criticality calculation, we give them here as a form of an interim report. For the various analyses of the experiment being conducted at KUCA, a relatively simple form of constants are needed, which is the reason for this presentation. Because of its complicated fuel-region geometry, repeated spectrum calculations are required. The report spends a large portion to describe such procedure.

I. Introduction

A cylindrical core loaded with 45%-enriched uranium reached its first criticality on May 12, 1981,[1,2] in the C-core of Kyoto University Critical Assembly (KUCA). This loading has a single cylindrical core at the center, with an annular D_2O tank as a reflector on its circumference and has axial symmetry (Fig. 1). The experiment was carried out on the initiative of the Research Reactor Institute, Kyoto University, with the assistance of the reactor physics group of Nagoya University, School of Engineering, as joint research participants. It was intended as the first step of the series of experiments that simulate the proposed high flux research reactor.[3] The latter reactor, although still in its proposal stage as of May, 1982,

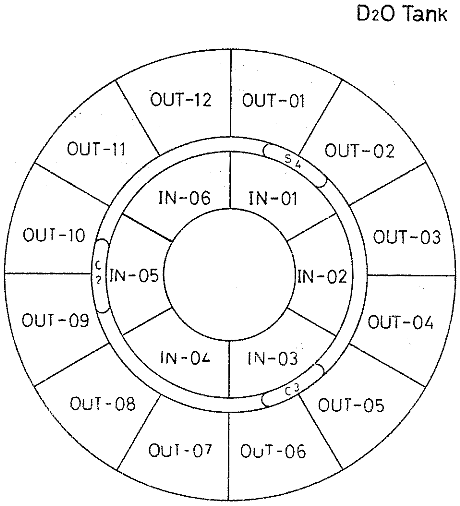


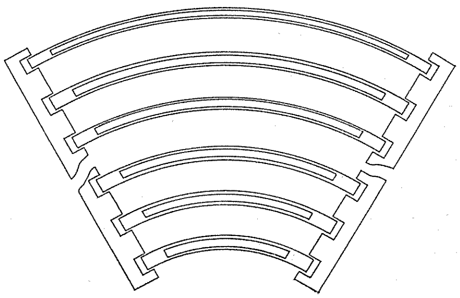
Fig. 1. The MEU-Core Loading in KUCA.
IN-01 ~ IN-06: Inner Fuel Assembly; OUT-01 ~ OUT-12: Outer Fuel Assembly; C2, C3: Control Rods #2 and #3; S4: Safety Rod #4.
In addition to these rods, one control rod (C1), two safety rods (S5 and S6) and six neutron detectors are allocated within light-water region outside the D_2O tank.

is scheduled to be loaded with such 45%-enriched uranium (called MEU in the following) fuel from the second operation year on. Such fuel loading program has been laid out so that it conforms with the nuclear non-proliferation policy of the United States announced by President Carter in April, 1977.[4)]

In this report the production of the reactor physics constants for the system will be described. These constants are necessary for the analysis of the criticality experiments mentioned above, and a series of experiments that were carried out afterwards by the Institute and other member universities of the joint program. Although such constants have already been prepared and reported by the members of the Research Reactor Institute,[5)] it is meaningful to have an independent work, judging from the novelty of the fuel plate geometry, and the importance of this criticality experiment in Japanese research reactor program. The present calculation employs different energy group structures and different region assignment pattern from those in Ref. 5).

In contrast to the rectangular core of the KUCA loaded with flat fuel plates, the present core has a complicated geometry that requires special care in reactor constant production. The core is loaded with concentric fuel plates, which are grouped by units of fan-shaped segments (Figs. 1 and 2). For this geometry our conventional scheme[6),7)] that relied on UGMG[8)] and THERMOS[9)] codes are simply

Fig. 2. A Schematic Figure of an MEU Fuel Assembly.
Seventeen fuel plates are loaded in the outer fuel region assembly, and fifteen in the inner fuel region assembly. The width of the plate varies depending on the position of insertion. These assemblies are called "Baumkuchen" type and abbreviated as "BK" for its similarity to the cake.



inadequate. UGMG performs homogeneous medium calculation, or heterogeneous medium calculation for rod type fuel, while THERMOS carries out one-dimensional transport theory calculation. The geometry indicated in Figs. 1 and 2 cannot be treated by such simple models. Moreover, due to the insufficient editing function of these codes, repeated spectrum calculations cannot be performed.

For this reason we employed MGCL-PROCESSOR system,[10] which was made available by the courtesy of Mr. Y. Naito, Japan Atomic Energy Research Institute. This system, which has been developed for criticality safety study, is a collection of many computer programs. It consists of microscopic cross section libraries, several editing programs, a transport equation program, Monte Carlo programs, diffusion theory programs, and evaluation programs. In the present work, however, we use only the following combination:

(a) Averaged microscopic cross section libraries "MGCL26GR"[10] and "SMF-26GR."[10]
(b) Program "MAIL"[10] that reads (a), and generates input macroscopic cross sections for "ANISN-JR."[11]
(c) One-dimensional transport equation code "ANISN-JR."
(d) Program "REMAIL"[10] that edits the output data of ANISN-JR and prepares macroscopic input data for the next run of ANISN-JR.

The procedure of adapting the critical assembly to the code system, and the data flow through the system will be described in the following chapters.

II. The Description of the Experiment

2. 1. The Experimental Apparatus[5,12]

The cylindrical core of the system has two concentric rings of fuel region, with a ring of water channel in between (Fig. 1). The inner ring, called inner fuel region, consists of six fan-shaped fuel assemblies (IN-01 through IN-06, Fig. 1), whereas the outer ring consists of twelve (OUT-01 through OUT-12). The ring of water channel between these fuel regions is spared for the insertion of three rods, denoted C2, C3, and S4, respectively, in the figure. Inside the inner fuel region is a center water island. The whole system described thus far is immersed within a cylindrical light-water space enclosed by the inner wall of a heavy-water reflector tank.

This tank[13] is a cylindrical shell made of aluminum, with the inner radius of 22.50cm and the outer radius of 54.00cm. The outer wall of the tank is 1.0cm thick, while the inner wall 0.5cm thick, leaving inside a heavy-water layer of 30.00cm thickness. Beyond the outer wall of the tank is light-water, which is thicker than 30cm, but replaced by 30cm of light-water in the present calculation, which is sufficiently thick to be considered as an infinite reflector.

The system is controlled by three control rods (C1, C2, and C3), and three safety rods (S4, S5, and S6). Out of the six rods, three (C2, C3, and S4) are placed inside the core as previously mentioned, and the rest are placed within the light-water region outside the heavy-water tank.

One of the fan-shaped fuel assemblies is illustrated in Fig. 2. An assembly for the inner fuel region can hold a maximum of fifteen curved fuel plates, while the one for outer region, seventeen. Each curved fuel plate is 1.4mm thick and

650mm long, with the width varying from plate to plate. Thus the widths of the plates for an inner fuel assembly range from 48.70mm to 104.41mm, while for an outer assembly from 61.16mm to 93.00mm, corresponding to the different length of arcs in the fan.

Each fuel plate holds inside a layer of uranium-aluminum alloy called "meat," which is 0.5mm thick and 600mm long, and clad with aluminum. For an inner assemblies the width of the meat varies from 39.50mm to 95.21mm, while for an outer assembly from 51.96mm to 83.80mm, corresponding to the different width mentioned above for different plates. The uranium-aluminum alloy of the meat contains 42 weight per cent of uranium whose ^{235}U content is enriched to 45%. The name "medium enriched uranium (MEU)" originates from this much enrichment. The curved fuel plates are inserted to a fuel assembly frame with 3.84mm pitch. The detailed specifications of these fuel plates are given in Ref. 5).

2. 2. Criticality Experiment and Critical Mass[2,12]

In the criticality experiment, fuel plates were loaded starting from the outermost position and proceeding inward toward the center of the cylindrical core. The criticality was reached when 58 plates were loaded in the inner fuel region, with all the outer fuel assemblies fully loaded ($17 \times 12 = 204$ plates) and one of the three control rods inserted partially into the core. The reactivity worth of this partial insertion was determined by a period measurement to be $0.211\% \Delta k/k$ while the worth of the 263rd plate (extra plate) was $0.263\% \Delta k/k$. Therefore by a linear interpolation the excess reactivity held by the core of 262 plates is estimated to be equivalent to 0.80 times the reactivity of a single plate at the latest loading position. The number of fuel plates corresponding to this minimum criticality is therefore 261.2. On the average then a single inner assembly holds 9.533 plates. The minimum critical mass is evaluated from specifications of the fuel plates[5] to be 4.155kg of ^{235}U.

III. Production of Group Constants

Prior to the ANISN-JR calculations in which neutron spectrum and weighted cross sections are evaluated, two steps of input data preparations are necessary, namely

(1) To devise a modelled system from the real experimental system,

(2) Determination of the number density of each isotope in each region.

We will first describe these two steps in Secs. 3-1 and 3-2; then explain the calculational flow in 3-3, and each step of the flow in 3-4 and 3-5.

3. 1. Modelling of the System

(a) The eleven region boundaries for the whole cylindrical system

We regard the system of Fig. 1 as one-dimensional cylindrical geometry consisting of eleven regions as shown in Fig. 3, including reflector regions of heavy-water and light-water outside. The radii of respective boundaries are known, with the exception of those for the fuel regions. The latter needs careful arithmetic, which we are to sketch briefly.

Inner fuel region: While the thickness of a fuel plate is 1.4mm, the width of a

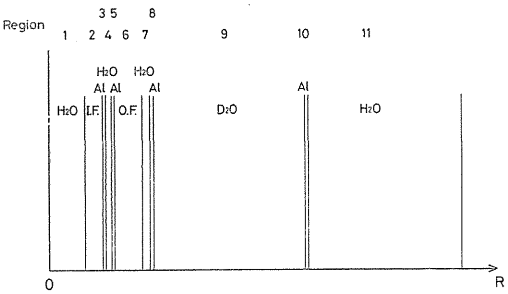


Fig. 3. The Cylindrical MEU-Core System in KUCA with Eleven Regions.
H_2O : Light-water region;
D_2O : Heavy-water region;
Al: Aluminum region;
I. F.: Inner fuel region;
O. F.: Outer fuel region.

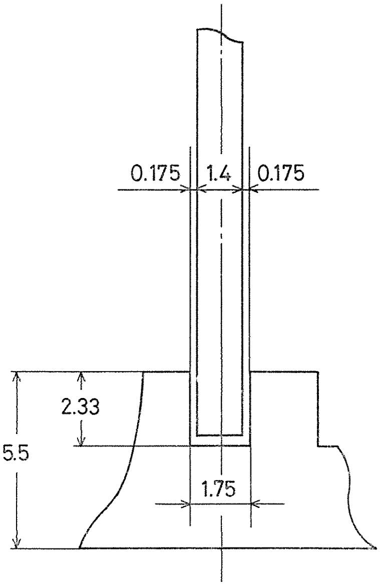


Fig. 4. The Relative Position of a Fuel Plate with Respect to the Groove on a Side-plate of an Assembly.

frame groove to which the plate is inserted is 1.75mm (Fig. 4). We assume that their center surfaces coincide. The unit cell is taken so that it has the same dimension as the fuel plate pitch (3.84mm). The cell consists then of a fuel plate (1.4mm thick), and water layers (1.22mm thick) on both sides. The number of fuel plates are taken to be 9.533, as mentioned. Furthermore, the distance between the last (or first) groove of the frame and the edge of the frame in Fig. 2 is taken into account. Based on these considerations the values 75.70mm and 114.00mm

Table 1. Materials and Outer Boundary Radii of Eleven Regions in Figure 3.

Region	Material	Radius of the Outer Boundary (cm)	Remarks
1	H_2O	7.57	Center gap region
2	Fuel	11.40	Inner fuel region
3	Al	11.65	
4	H_2O	12.85	Control rod space
5	Al	13.10	
6	Fuel	19.78	Outer fuel region
7	H_2O	22.50	
8	Al	23.00	Inner wall of the D_2O tank
9	D_2O	53.00	Heavy-water reflector region
10	Al	54.00	Outer wall of the D_2O tank
11	H_2O	84.00	Light-water reflector region

are assigned to the inner and outer radius of this fuel region, respectively.

Outer fuel region: By similar considerations, the radii 131.00mm and 197.80mm are assigned for the inner and outer boundaries.

Table 1 summarizes the radii assigned to the outer boundaries of the eleven regions (Fig. 3). The height of the fuel region is taken equal to that of the meat layer, 600mm.

(b) Treatment of a single fan-shaped fuel assembly

ANISN-JR cannot solve a problem of a fan-shaped fuel region illustrated in Fig. 2. We thus replace this region (Fig. 5(a)) by an equivalent slab system of the same area (Fig. 5(b)). This equivalent system has the following six regions:

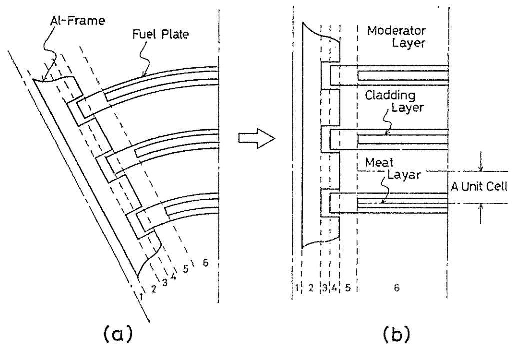


Fig. 5. Modelling of a Fuel Assembly. Fan-shaped assembly (a) is deformed to a rectangular assembly (b) by conserving the area of the assembly.

♯1: A narrow light-water region found between the adjacent frame when the core of Fig. 1 is constructed.

♯2: The side plate of the fuel frame excluding the groove portion.

♯3: Light-water filling the groove plus aluminum frame.

♯4: Similar to the region ♯3, but the light-water replaced partially by aluminum of a fuel plate.

♯5: Aluminum cladding and light-water.

♯6: Consists of the meat, aluminum cladding and light-water.

3. 2. Number Density Determination

The number density of light-water is based on its mass density at 20.44℃.[14] For all other materials, the values at room temperature are employed.[15] The enrichment of heavy-water is put 99.45 per cent (the rest is light-water).

To describe the procedures for the six regions of Fig. 5(b), the regions ♯1 and ♯2 are treated as made of single materials. For the regions ♯3 through ♯5, aluminum and light-water are mixed uniformly within each region. The macroscopic cross sectios with these number densities are then directly edited by the program MAIL.

For the region ♯6, a different scheme is employed, corresponding to an extra step needed in the cross section averaging. A one-dimensional, three-region problem is solved first by ANISN-JR in the direction perpendicular to the fuel plate, a step not needed for the previous five regions. For this step number densities of three layers, namely the layers of light-water, aluminum cladding and the meat, are needed. For the first two, pure material is assumed, while for the meat layer a

Table 2. Number Densities of Isotopes in Fuel Region.
The region numbers are those indicated in Fig. 5.

($\times 10^{24}$ n/cm^3)

Region			H	O	Al	^{235}U	^{238}U
♯1			0.066734	0.033367			
♯2					0.060244		
♯3			0.030412	0.015206	0.032789		
♯4			0.0060826	0.0030413	0.054753		
♯5			0.042404	0.021202	0.021964		
♯6	Moderator Region		0.066734	0.033367			
	Cladding Region				0.060244		
	Meat Region	Inner Fuel			0.052858	0.0019444	0.0023639
		Outer Fuel			0.052461	0.0019342	0.0023453

Table 3. Number Densities of Isotopes in the Al, D_2O and H_2O Regions.

($\times 10^{24}$ n/cm^3)

Region	H	D	O	Al
Al				0.060244
D_2O	0.00036704	0.066098	0.033233	
H_2O	0.066734		0.033367	

uniform mixture of ^{235}U, ^{238}U and aluminum is employed.

The number densities thus determined for the fuel regions (those in Fig. 5(b)) are tabulated in Table 2. Those for nonfuel regions (regions 1, 3, 4, 5, 7, 8, 9, 10, and 11 of Fig. 3) are tabulated in Table 3.

After the ANISN-JR calculation described above for the region ♯6, the macroscopic cross sections of this region are edited by MAIL, and used together with those of the regions ♯1 through ♯5 mentioned already. Then a six-region problem is solved in the direction parallel to the fuel plate. The flow of such calculation is described in detail in the next section.

3. 3. The Flow of Calculation

Before one obtains the final constants for the one-dimensional cylindrical system of Fig. 3, one has to go through several steps, in which a heterogeneous system of each stage is homogenized with neutron flux as weighting factors. The energy- and space-dependent neutron flux necessary in each step for such cross section averaging are all calculated by ANISN-JR code in the present work. Such flow of calculation is illustrated in Fig. 6. In the figure there are two kinds of jobs connected to each other. They are (1) preparation of input files (macroscopic cross sections) for ANISN-JR, and (2) calculations by ANISN-JR. In describing these steps in detail, we place emphasis in the procedure for the inner core, the one for the outer core being quite similar.

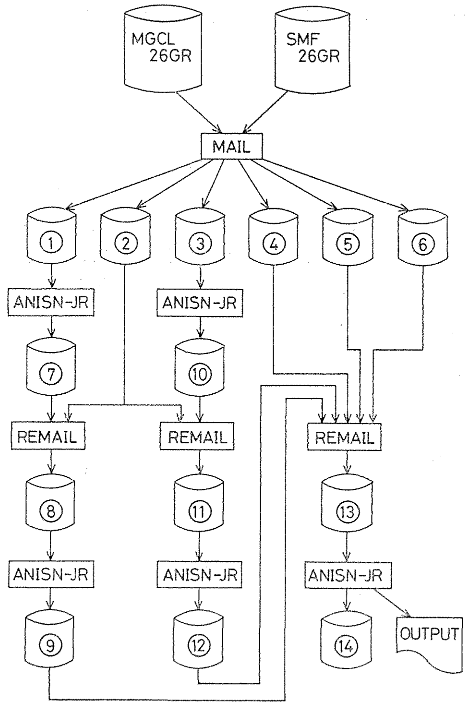


Fig. 6. The Flow of Preparing Two-group Constants.

Step ♯1: By the use of program MAIL, the three-region input file ① for ANISN-JR is prepared for the region ♯6 of Fig. 5(b).

Step ♯2: By MAIL, the five-region input file ② for ANISN-JR is prepared, corresponding to the regions ♯1 through ♯5 of Fig. 5(b).

Step ♯3: By MAIL, the input files ④ through ⑥ are prepared for ANISN-JR, each file corresponding to aluminum, heavy-water, and light-water region, respectively.

Step ♯4: ANISN-JR solves a three-region problem of the region ♯6, Fig. 5(b). With this output flux as weighting factors, the region is homogenized, and the macroscopic cross section file ⑦ is prepared.

Step ♯5: REMAIL combines the files ② (for the regions ♯1 through ♯5 of Fig. 5(b)) and ⑦ (for the region ♯6 of Fig. 5(b)) to make one file ⑧.

Step ♯6: ANISN-JR solves the six-region problem of the regions ♯1 through ♯6 in Fig. 5(b). With this output flux as weighting factors, the region is homogenized, to form the region ♯2 of Fig. 3, and the macroscopic cross section file ⑨ is prepared.

Step ♯7: One prepares the macroscopic cross section file ⑫ for the outer core. The order of preparation: ⑩ is obtained from ③, and combined with ② to form ⑪, from which ANISN-JR prepares ⑫. It is the same scheme as the sequence of Steps ♯1, ♯2, ♯4, ♯5, and ♯6 for the inner core.

Step ♯8: REMAIL combines the files ⑨ (inner fuel region), ⑫ (outer fuel region), ④ (aluminum), ⑤ (heavy–water), and ⑥ (light-water) to prepare the eleven-region macroscopic cross section file ⑬ for the final run (criticality calculation) of ANISN-JR. The eleven regions are those indicated in Fig. 3.

Step ♯9: With input file ⑬, ANISN-JR solves the whole cylindrical system as an eleven-region problem of Fig. 3, and prepares two-group constants by taking averages with this output flux. The output is made on the file ⑭ and on the print-out.

3. 4. Multi-group Libraries, and Input Data Preparation for ANISN-JR

(a) Libraries

Two multi-group libraries are employed in the present calculation (Fig. 6), namely MGCL26GR and SMF26GR. These libraries as a whale play similar roles as JAERI Fast Set,[16] or ABBN Set.[17] MGCL26GR is a one-dimensional set of twenty-six-group constants obtained by collapsing 137-group constants, the latter having been prepared with MGCL-ACE[10] code system from nuclear data libraries ENDF/B-IV[18] and JENDL.[19] MGCL26GR consists of total cross section σ_t, absorption cross section σ_a, fission cross section σ_f, average number ν of neutrons emitted per fission, elastic scattering cross section σ_e, inelastic scattering cross section σ_{in}, the cross section $\sigma_{n,2n}$ for (n, 2n) reactions, self-shielding factor f, mutual shielding factor g, and correction factor h for moderator mass effect. SMF26GR is a two-dimensional set consisting of elastic and inelastic scattering matrices, and (n, 2n) reaction matrix.

(b) Preparing macroscopic cross sections from the libraries MGCL26GR and SMF26GR

MAIL program prepares macroscopic cross sections for ANISN-JR from the two libraries MGCL26GR and SMF26GR (Fig. 6). By input parameters of MAIL, one specifies the number of regions, number of nuclides, nuclide name, and number density of each nuclide. For fuel regions, parameters for Dancoff correction are also needed.

(c) Editing macroscopic cross section sets

In the present calculation, the fuel region requires extra steps of homogenization (Sec. 3–3), and consequently several sets of cross sections are prepared as input for the final criticality calculation (Step ♯9). ANISN-JR can, however, handle only a single input library file, and thus the program REMAIL (Fig. 6) combines several sets of macroscopic cross sections to form a single set.

3. 5. One-dimensional Transport Equation Calculation for the Preparation of the Constants

Three kinds of problems are solved by ANISN-JR code, as already outlined in Sec. 3–3 and Fig. 6. All of the calculations are carried out with P_1-approximation with regard to the neutron anisotropic scattering. Angular quadrature sets with four abscissas are employed (S_4-calculation) for the integration with respect to the solid angle, which means that five representative directions of neutron motion are employed in the calculation.

(a) Three-region calculation including the meat (Step ♯4)

The equation for the region ♯6 of Fig. 5(b) is solved in the direction perpendicular to the fuel plate. The space between the center plane of the meat layer and the center of the moderator gap is taken as a unit cell, and treated as a plane geometry problem. Reflective boundary conditions are employed for both boundaries.

After the run, the constants are reduced to a set of one-region constants, with twenty-six energy groups retained.

(b) The six-region calculation for the inner fuel region (Step ♯6), and the outer fuel region

The equation for the six-region plane geometry of Fig. 5(b) is solved in the direction parallel to the fuel plates. Reflective boundary conditions are applied on the outer edges of the regions ♯1 and ♯6. After the run, a set of one-region, twenty-six energy-group constants are prepared.

(c) Criticality calculation of the whole system (Step ♯9)

The cylindrical system of eleven regions in Fig. 3 is solved with twenty-six energy-group structure and with the following boundary conditions: albedo is unity at the center for all energy groups, and zero at R_s, where R_s is the outer radius of the region 11. The multiplication factor (eigenvalue) k determined was 1.0055.

Table 4. Fast Group Constants of the MEU System.

Region	Material	D (cm)	Σ_a (cm^{-1})	$\nu\Sigma_f$ (cm^{-1})	Σ_R (cm^{-1})
1	H_2O	1.0254	6.1859×10^{-4}	0.0	5.2656×10^{-2}
2	Inner Fuel	1.4242	3.9959×10^{-3}	4.3095×10^{-3}	2.3341×10^{-2}
3	Al	2.9478	6.8406×10^{-4}	0.0	2.0368×10^{-4}
4	H_2O	1.0453	5.3938×10^{-4}	0.0	4.2859×10^{-2}
5	Al	2.9471	6.7898×10^{-4}	0.0	2.0350×10^{-4}
6	Outer Fuel	1.4310	3.9558×10^{-3}	4.2561×10^{-3}	2.3090×10^{-2}
7	H_2O	0.98418	5.8804×10^{-4}	0.0	5.3728×10^{-2}
8	Al	3.0613	6.7867×10^{-4}	0.0	2.8358×10^{-4}
9	D_2O	1.3491	1.6821×10^{-4}	0.0	1.1035×10^{-2}
10	Al	3.7905	1.1758×10^{-3}	0.0	7.2368×10^{-4}
11	H_2O	1.0177	1.3691×10^{-3}	0.0	1.6652×10^{-1}

Table 5. Thermal Group Constants of the MEU System.

Region	Material	D (cm)	Σ_a (cm^{-1})	$\nu\Sigma_f$ (cm^{-1})
1	H_2O	0.14404	1.7879×10^{-2}	0.0
2	Inner Fuel	0.25952	1.0650×10^{-1}	0.19107
3	Al	4.2113	1.0289×10^{-2}	0.0
4	H_2O	0.15396	1.6498×10^{-2}	0.0
5	Al	4.2115	1.0229×10^{-2}	0.0
6	Outer Fuel	0.26941	1.0226×10^{-1}	0.18384
7	H_2O	0.14622	1.7559×10^{-2}	0.0
8	Al	4.2094	1.1043×10^{-2}	0.0
9	D_2O	0.83558	1.2893×10^{-4}	0.0
10	Al	4.2085	1.1458×10^{-2}	0.0
11	H_2O	0.14064	1.8397×10^{-2}	0.0

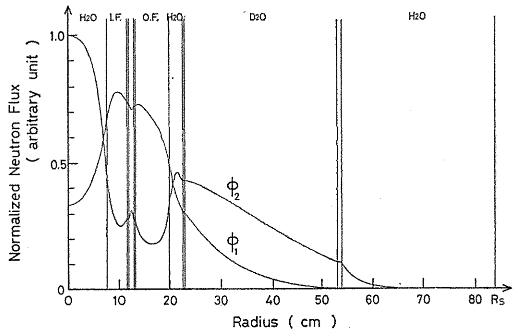


Fig. 7. Neutron Flux Distribution in Radialdirection by ANISN-JR.
ϕ_1: Fast neutron flux; ϕ_2: Thermal neutron flux.
The neutron flux distributions are normalized so that ϕ_2 at the center is unity.

With the output of this run, two-energy-group constants are generated for all eleven regions, which are tabulated in Tables 4 and 5. The two-group flux as the sum of twenty-six group flux are illustrated in Fig. 7.

IV. One-Dimensional, Two-Group Criticality Calculation by Diffusion Theory Code

To check the feasibility of the two-group constants generated in Sec. 3–5(c), criticality calculation by one-dimensional diffusion code EXPANDA-25-IMPORT was performed. This code is a modified version of EXPANDA-25[20] with additional capability to calculate adjoint flux. The eleven-region scheme of Fig. 3 are employed with the z-direction dimension of $H+2\delta_z$, where H is the height of the meat region, 60.0cm. For δ_z, the measured reflector saving of the C–35 loading[21] with high-enriched uranium is borrowed, which is 8.2cm.

The boundary conditions applied are $\phi'(0)=0$ and $\phi(R_s)=0$. With a convergence criteria ε set equal to 1.0×10^{-5}, the eigenvalue k was 1.0436. The neutron flux and adjoint flux obtained are shown in Figs. 8 and 9, respectively. The flux are normalized so that the thermal flux is unity at the center, while adjoint flux are normalized so that the maximum of thermal adjoint flux is unity.

In comparing Fig. 7 and Fig. 8, we see the similarity of the thermal flux (ϕ_2) in both calculations. As to the fast flux (ϕ_1), the transport theory gives slightly

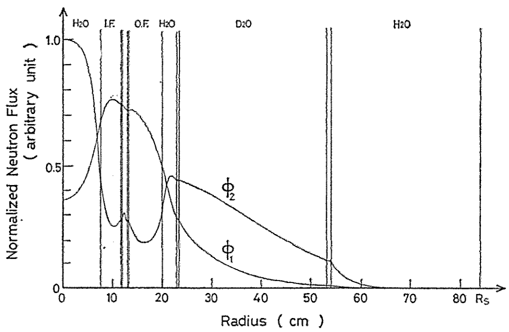


Fig. 8. Neutron Flux Distribution in Radial-direction by EXPANDA-25-IMPORT.
ϕ_1: Fast neutron flux;
ϕ_2: Thermal neutron flux.
The neutron fluxes are normalized so that ϕ_2 at the center is unity.

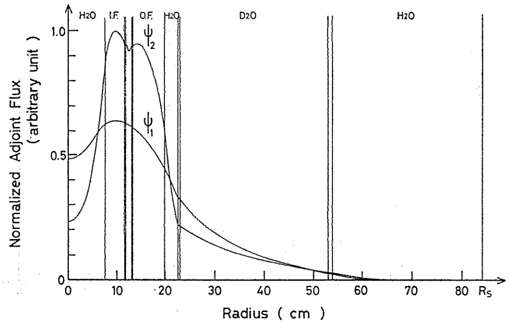


Fig. 9. Adjoint Flux Distribution in Radial-direction by EXPANDA-25-IMPORT.
ψ_1: Fast adjoint flux;
ψ_2: Thermal adjoint flux.
The adjoint fluxes are normalized so that the maximum value of ψ_2 is unity.

lower values in the center light-water region, while higher values in the fuel regions and in the heavy-water region. The overall distributions of these flux are, however, quite similar.

V. Discussions and Possibilities for Improvements

Between the values by ANISN-JR ($k=1.0055$) and EXPANDA-25-IMPORT ($k=1.0436$), there is a difference of 3.8%. The first possible source of this discrepancy is an inadequate δ_z in the latter calculation. The sensitivity of k on δ_z is, however, relatively small. If the present δ_z value of 8.2 cm is replaced by 7.7 cm, the resulting k becomes 1.0413, a shift of 0.23%. Therefore this cannot be the only source.

The second possibility is inadequate values of diffusion constants D used in the diffusion calculation. While all other constants for the diffusion calculation are taken directly from the outputs of ANISN-JR, there is a certain arbitrariness[22] in the choice of D. In the present calculation, the two-group constants D_1 and D_2 determined by

$$D_i=1/[3\Sigma_{si}(1-\bar{\mu})] \qquad i=1,\ 2$$

are used, where Σ_{si} are the scattering cross section within the i-th group evaluated by ANISN-JR.

This scheme of D-determination, however, has to be verified by the following test calculation, for example. Instead of collapsing the twenty-six-group constants directly to a two-group set, take an intermediate step of a four-group set. We then calculated four diffusion constants, one for each group, from the output on ANISN-JR calculation. Four-group diffusion calculation can be carried out with these constants to obtain the eigenvalue, and compare it with its two-group version. This is the first check.

Secondly, collapse four-group constants given by ANISN-JR to two-group constants by a desk calculator. This requires a careful treatment of ANISN-JR output with regard to P_0 and P_1 component of scattering cross section, and group-dependent average cosine of scattering angle. If the result is consistent with the two-group constants given in this report, then we can be sure that we have made valid editing of ANISN-JR output in obtaining the diffusion constant. At this stage we have not made complete understanding on the input and output format of ANISN-JR yet, and these checks have to be made.

Furthermore, the eigenvalues obtained in the spectrum calculations of Steps ♯4 and ♯6 are off unity in the present calculation. In future these values have to be adjusted by proper transverse buckling, so that even at the stage of the spectrum evaluation the system is critical.[23] In addition, possibility of energy-dependent δ_z must be examined, as the additional parameters of two-group constants. These parameters can be determined once a criticality calculation in $(r\text{-}z)$ geometry is made. Only after these improvements we would be ready to see if the eigenvalue still stays off unity, leaving discrepancy with experiment.

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