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ORR REACTOR PHYSICS: PROBLEM I,
INFINITE SLAB MODEL

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ORR REACTOR PHYSICS: PROBLEM I, INFINITE SLAB MODEL

1. Description of Physical System

The ORNL Research Reactor (ORR) is a light-water moderated and cooled system, fueled with highly-enriched uranium (greater than 90% isotopic concentration of U-235). A horizontal cross-section through the core of the ORR at the mid-plane is shown in Fig. 1.

In many respects the chain-reacting core resembles that of the Materials Testing Reactor (MTR), which is described in references (1), (2) and (3). The nuclear characteristics of the fuel elements proposed for the ORR probably will resemble closely those currently used in the MTR. The principal differences between the nuclear characteristics of the two systems result from their respective side reflectors. In the MTR, the side reflector is, in effect, an almost infinitely-thick beryllium layer whose nuclear characteristics are merely perturbed by the beam holes which penetrate it. The beryllium reflector configuration of the ORR shown in Fig. 1 is much thinner than an "infinite" beryllium reflector. Furthermore, except for a relatively unimportant (neutron-wise) layer of aluminum, the ORR beryllium is backed on three sides almost completely by voids (the beam holes and the large experimental facilities).

The chain-reacting core of the ORR is contained within a vertical aluminum housing which in turn is supported by the reactor tank. At the bottom of this housing is the seven-by-nine grid which holds the lower ends of both the fuel elements and of the beryllium reflector pieces. This grid will be fabricated from 303 stainless steel. The relationship of these and other components is shown in Fig. 2.

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FIG-1
HORIZONTAL MIDPLANE OF ORR CORE
(SEE ORR DWG. D-19719
FOR CONSTRUCTION DIMENSIONS)

As in the MTR, above and below the core an effectively-infinite water reflector is provided by the circulating moderator-coolant. Outside the housing, both large experimental facilities of the ORR are presumed to be empty, as are the beam holes which "see" almost all the central half of the core on one side. The large pool facility is taken to be filled with water.

Control of the chain-reaction is by vertical motion of control elements driven from beneath the reactor with shafts which penetrate the bottom plug of the reactor pool. As in the MTR, each control element of the ORR has both a poison-containing region and fuel-bearing region. As the poisoned portion of the control element is displaced, it is followed by the fueled portion, which compounds the reactivity effect as compared with simple poison or fuel displacement. The relationship of both portions of the control elements to the remainder of the core is included in Figure 2.

2. Description of Idealized System:

A primary objective of this initial calculation was to obtain as quickly as possible a rough idea of the neutron flux distribution. Toward this end, a highly simplified model was chosen, and some of the nuclear constants used were taken to be identical with values for a somewhat comparable fuel loading in the MTR.

The flux distribution function was postulated to be separable in the three coordinate directions; i.e., the flux at a point could be calculated from an expression of the form $\phi(\underline{r}) = \phi_1(x) \phi_2(y) \phi_3(z)$. The flux distributions along each coordinate were obtained from a two-group, three-region analysis of an infinite slab. The transverse configuration of each slab was chosen to correspond to that along one coordinate direction

FIG.-2
VERTICAL SECTION Z-Z THRU ORR CORE

in the actual core. The dimensions of each slab are compiled in Table 3.1 and shown also in Figure 3. No attempt was made to include the effect of the finite dimensions of the reactor in the directions perpendicular to the one being considered.

The properties of the fuel-bearing region used in each case are presented in Table 3.2. They were chosen to be identical with those given by Holm et al⁽⁴⁾ for the MTR with 168-gram 18-plate fuel elements and 131-gram 14-plate shim-fuel elements, arranged in a three-by-nine loading composed of twenty-three fuel elements and four shim-fuel elements. For comparison, the ORR configuration shown in Fig. 1 is a four-by-seven array with 24 fuel elements and four shim-fuel elements.

The core nuclear constants, for a homogeneous mixture of the constituents computed for an operating temperature of 300°K, are compiled in Table 3.2. The cross sections were averaged over a Maxwellian distribution of neutron energies with a corresponding mean value. Half the water between the fuel elements and the reflector was arbitrarily assigned to each of the two regions. The Al/H₂O volume ratio in the core was 0.732.

The fuel-containing region of each slab was encased in a symmetrical two-region reflector in the x- and y-directions. The inner reflector is an homogenized mixture of the beryllium and water of the reflector, while the outer reflector is composed of 1.25 inches of aluminum. The z-direction reflector zones are respectively composed of an homogenized mixture of the metals and the water adjacent to the fueled region, and a 50-cm thick layer of ordinary water. The reflector constants, which were also chosen at 300°K are presented in Tables 3.3, 3.4, 3.5, and 3.6.

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FIG.-3
IDEALIZED SLAB REACTORS
FOR VARIOUS COORDINATE DIRECTIONS

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Symmetrization of the inner reflector regions in the y- and z-direction was realized in each case by computing the amount of material present on both sides of the actual core and arbitrarily assigning half the total to either side of the idealized slab.

3. Calculation Procedures:

The numerical computations were made by the ORACLE using the three-group three-region code given by Bate, Einstein, Coveyou and Kinney⁽⁵⁾ which has provision for specializing to a two-group computation. That is, the following differential equations were solved numerically:

$$\Phi_{ij}''(x) - \alpha_{ij} \Phi_{ij}(x) + T_{ij}(x) = 0 \quad (3.1)$$

$$D_{ij} T_{ij}(x) = f_{2j} \Phi_{2j}(x) \quad (3.2)$$

$$D_{2j} T_{2j} = \sum_{1j}^x \Phi_{1j}(x) \quad (3.3)$$

i = 1, 2 denotes neutron-energy group: fast and thermal, respectively.

j = 1, 2, 3 denotes region: core, first reflecting region, and second reflecting region, respectively.

D ≡ diffusion coefficient, cm

Φ ≡ flux neutrons/(cm²) (sec)

f ≡ fission cross section (per cm)

Σ^a ≡ absorption cross section (per cm)

Σ_i^x ≡ transfer cross section from group i
to group (i + 1) (per cm)

Dα ≡ Σ^a + Σ^x - B² D (per cm)

B² ≡ unreflected buckling per sq. cm.

f ≡ ν Σ^f per cm

ν ≡ number of neutrons per fission

The transfer cross section Σ_1^x from group 1 (fast) to group 2 (thermal) is given by

$$\Sigma_{1j}^x = D_{1j} / \tau_j^2$$

τ_j = Fermi age to thermal energy for prompt fission neutrons in the material, sq. cm.

The boundary conditions are:

- (i) symmetry of flux with respect to center plane of the slab
- (ii) continuity of flux at interface of two regions
- (iii) continuity of current at interface of two regions
- (iv) return current is zero at outer bounding plane of reactor.

TABLE 3.1

Dimensions (cm) of Various Regions in Each Slab*

	x	y	z
t_1	27.70	16.15	30
t_2	11.62	7.75	18.73
t_3	3.175	3.175	50

TABLE 3.2

Core constants⁽⁴⁾: 168-Gram 18-Plate Fuel Elements and 131-Gram 14-Plate Shim-Fuel Elements at 300° K

	Fast	Thermal	
Σ^a	0	0.07992	cm ⁻¹
D	1.124	0.269	cm
$\nu \Sigma^f$	0	0.1340	cm ⁻¹
$\nu \Sigma^x$	0.01756	0	cm ⁻¹
τ	64	—	cm ²
L^2	—	3.366	cm ²
B^2	0	0	cm ⁻²

*See Fig. No. 3

TABLE 3.3

Constants for Be-H₂O Reflector Regions

	<u>x-direction</u>		<u>y-direction</u>		
	93% Be and 7% H ₂ O		95% Be and 5% H ₂ O		
	Fast	Thermal	Fast	Thermal	
Σ^a	0	0.00233	0	0.000197	cm ⁻¹
D	0.604 ⁽¹⁾	0.699 ⁽²⁾	0.602 ⁽¹⁾	0.765 ⁽²⁾	cm
Σ^f	0	0	0	0	cm ⁻¹
Σ^x	0.007744	0	0.007253	0	cm ⁻¹
τ	78 ⁽³⁾	-----	83 ⁽³⁾		cm ²
B ²	0	0			

(1) By linear extrapolation from values reported in IDO-16082⁽⁴⁾
for 0% - 100%, 2% - 98% and 3% - 97%

(2) Second method, page 15, IDO-16082⁽⁴⁾

(3) From a quintic parabola through values reported in IDO-16082⁽⁴⁾
and 33 cm² for water.

TABLE 3.4

Constants for Aluminum Reflector Regions

	Fast	Thermal	
Σ^a	0	0.0127	cm ⁻¹
D	3.542	2.032	cm
Σ^x	1.322x10 ⁻⁴	0	cm ⁻¹
Σ^f	0	0	cm ⁻¹
τ	26800	-----	cm ²
B ²	0	0	cm ²

TABLE 3.5

Constants for Al-H₂O-Stainless Steel Reflector Regions

	Fast	Thermal	
Σ^a	0	0.02335	cm ⁻¹
D	1.27	0.233	cm
Σ^f	0	0	cm ⁻¹
Σ^x	0.0233	0	cm ⁻¹
τ	54.4	-----	cm ²
B ²	0	0	cm ⁻²

TABLE 3.6

Constants for H₂O Reflector Regions

Water @ 57.2°C

	Fast	Thermal	
Σ^a	0	0.01855	cm ⁻¹
D	1.166	0.1558	cm
Σ^f	0	0	cm ⁻¹
Σ^x	0.0340	0	cm ⁻¹
τ	34.3	-----	cm ²
B ²	0	0	cm ⁻²

4. Fluxes and Beam Current

The fluxes computed on the basis outlined above are plotted in figures 4, 5, and 6. Each thermal flux distribution plotted is normalized to unity at the center of the reactor. At a power level of 20 megawatts each of them must be multiplied by 5.69×10^4 . The product of the three flux functions is an estimate of the flux in the reactor which at 20 megawatts is 1.8×10^{14} at the reactor's center. Denoting this thermal flux at

FIG 4

NEUTRON FLUX DISTRIBUTIONS IN X-COORDINATE SLAB

TWO-NEUTRON-GROUP CALCULATION FOR FUEL-CONTAINING CORE AND TWO-REGION REFLECTOR IN INFINITE-SLAB GEOMETRY, CORRESPONDS TO SYMMETRIZED AND HOMOGENIZED 4x7 LOADING OF ORR CORE.

FIG. 5
NEUTRON FLUX DISTRIBUTIONS IN Y-COORDINATE SLAB
TWO-NEUTRON-GROUP CALCULATION FOR FUEL-CONTAINING CORE
AND TWO-REGION REFLECTOR IN INFINITE-SLAB GEOMETRY COR-
RESPONDS TO SYMMETRIZED AND HOMOGENIZED 4x7 LOADING OF
O R R CORE

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FIG. 6

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a point P by $\Phi(P)$ and recalling that the return current at the exterior faces of the reactor was taken to be zero we have

$$\Phi(P) = -2D\Phi_y(P) = 0.269 \times 10^{14} \text{ neutrons/cm}^2\text{-sec}$$

$$\Phi(R) = -2D\Phi_x(R) = 0.194 \times 10^{14} \text{ neutrons/cm}^2\text{-sec}$$

$$J(P) = 1.34 \times 10^{13} \text{ neutrons/cm}^2\text{-sec}$$

$$J(R) = 0.97 \times 10^{13} \text{ neutrons/cm}^2\text{-sec}$$

where J is the current leaving the reactor.

Using the expression

$$j(u; P) = \frac{1}{4\pi} \left\{ \Phi(P) \mu - 3D\Phi_y(P) \mu^2 \right\}, \quad 0 \leq \mu \leq 1 \quad (4.1)$$

for the angular distribution of the current (when μ is the cosine of the angle between the normal to the reactor at P and an arbitrary direction), the current at Q is

$$J(Q) = \int_0^{2\pi} d\Psi \int_{\mu_0}^1 j(\mu; P) d\mu = 5.0 \times 10^9 \quad (4.2)$$

where μ_0 is given by

$$\mu_0 = 14.5 \sqrt{\frac{1}{16} + 14.5} \quad (4.3)$$

Note that the beam current $J(Q)$ given above includes only those neutrons which reach Q without scattering, it being supposed that the current density in the neighborhood of P is that at P.

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FIG. 7

RELATIONSHIP OF VARIOUS QUANTITIES IN EXPRESSION
FOR BEAM-HOLE CURRENT, EQ. 4.1

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References

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1. Huffman, John R., "The Materials Testing Reactor Design", Reactor Science and Technology 2, No. 4 (Dec. 1952) TID-2004, (Secret).
2. Buck, J. H., and Leyse, C. F., (eds.), "Materials Testing Reactor Handbook", ORNL 963 (Secret).
3. "Three Declassified Reports on the MTR", ORNL CF-53-12-21, (Declassified).
4. Holm, M. W.; Webster, J. W.; and McMurry, H. L., "Nuclear Constants for the MTR with 168-Gram Fuel Elements and 131-gm Fuel-Bearing Shim Rods", IDO-16082, (May 8, 1953), (Secret).
5. McMurry, H. L., "Nuclear Constants for the MTR as a Function of Fuel Content, Poison Content, and Al: H₂O Ratio", IDO-16127, (Oct. 23, 1953) (Secret).
6. Bate, R. R.; Einstein, L. T.; and Kimney, W. E.; "Description and Operating Manual for the Three Group, Three Region Reactor Code for Oracle", ORNL-CF-55-1-76 (Unclassified).

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