CONF-850507--14

Neutronic Calculations for a New High Flux Reactor*

CONF-850507--14

DE85 011551

F. C. Difilippo
B. A. Worley
D. R. Vondy

Paper to be presented at the

International Conference on

Nuclear Data for Basic and Applied Science

Santa Fe, New Mexico
May 13-17, 1985

*Research sponsored by the Division of Materials Sciences, U.S. Department of Energy, under Contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

ABSTRACT

The Oak Ridge National Laboratory has begun the design of a new high flux reactor to be used for basic research, isotope production, and material irradiation. One of the principal goals of the design is the production of a thermal flux peak in the reflector larger than $5 \times 10^{15}/\mathrm{cm}^2\mathrm{sec}$. None of the existing steady sources of neutrons in the world produce such neutron flux. A theoretical analysis of the slowing-down and diffusion of neutrons produced by a spherical fission source immersed in a moderator shows that the flux per unit power is maximized by combining a very under-moderated core with a very low absorbing reflector. The theoretical model interrelates total power, power density and transport properties with the thermal flux allowing very inexpensive scoping calculations. Full scale and detailed calculations were made with a numerical model which uses the BOLD VENTURE code system. Calculations show that a highly enriched ^{235}U reactor with D_2O as moderator and reflector would produce the desired peak flux, and the reactor would have a reasonable core life.

I. INTRODUCTION

The purpose of this paper is to present preliminary calculations of a new high flux reactor to be used for basic research, isotope production and material irradiation. The reactor would fill a gap caused by the increasing use of neutron scattering experiments in diverse disciplines[1] and the natural aging of existing facilities. The goal is to design a steady source of neutrons capable of producing a thermal neutron flux, ϕ, in the reactor larger than $5 \times 10^{15}/\mathrm{cm}^2\mathrm{sec}$ with a desired operating period of two or more weeks between refueling. None of the existing steady sources of neutrons in the world produce such a neutron flux. One objective of the calculations is the maximization of the ratio ϕ /P where P is the power generated in the fission process. This objective is quite opposite to the more usual applications of a reactor as a source of energy.

II. PARAMETRIZATION OF THE DESIGN

The optimization of the design is related with the problem of slowing-down and diffusion of neutrons from a spherical fission source immersed in a moderator. This problem can be solved analytically allowing very inexpensive scoping calculations and the individualization of the more important parameters. The first step toward the solution is the calculation of the Green's functions:

$$\nabla^2 G_i - \frac{G_i}{L_i} + \frac{1}{D_i} \delta(r - r') = 0 \tag{1}$$

where G_i (r, r') are the two-region diffusion kernels corresponding to a shell source located at r'; index i is equal to 1 for the core and equal to 2 for the reflector, L_i and D_i are, respectively, the diffusion length and the diffusion constant of region i. The thermal fluxes are given by the equations:

$$\phi_i (r) = \int_0^\infty 4\pi r'^2 G_i(r,r')S(r')dr' \tag{2}$$

where S(r') is the slowing-down density evaluated at thermal energies. We have used Fermi age theory to evaluate S in terms of the ages of region i, τ_i, assuming a flat fission rate. In this way it was possible to find an explicit analytical form for the flux in terms of exponential and error functions. Table I (cases 2 to 5) shows an application of the model to the case of *spherical cores moderated and reflected* with D_2O using Al clad ^{235}U fuel elements. The parameters not appearing in Table I are L_2 = 184 cm, D_2 = 0.831 cm, τ_2 = 131 cm^2 (pure D_2O) and τ_1 = 357 cm^2 (Al + D_2O mixture[2]). Table I also shows results corresponding to the ideal case of a point source of fission neutrons, in all cases the sources were scaled according to the energy release per fission. The required powers to produce a specified flux predicted by the analytical model were compared with the predictions of the numerical model described in Section III. We found an excellent agreement (discrepancies around 1%). The flux-to-power ratio is then determined by the eight parameters L_1, D_1, τ_1, L_2, D_2, τ_2, $\underline{V}$ and $\underline{d}$. Table I (column F) shows that the combined effect of large power density (i.e. small core volumes) and large neutron ages (i.e. bad moderators) results in a large fraction of the fission neutrons being moderated in the reflector; these neutrons are trapped there provided we use a thick and low absorbing reflector. The idealized case of a point source gives the minimum power required to create a neutron field with a fission source.

III. FULL SCALE CALCULATIONS

The previous analysis shows that the ratio ϕ /P is maximized provided we design the reactor to minimize V, L_1 and L_2 and to maximize τ_1 and $\underline{d}$. One good combination for this purpose is the use of highly enriched ^{235}U fuel and heavy water as moderator and reflector. The detailed calculations include the effects of target, structural materials, control rods and burn-up and were performed using a 6 energy group, two-dimensional (r, Z) numerical model. The diffusion equations were solved with the BOLD VENTURE code system.[3]

The calculational procedure was tested by modeling HFIR[4] (High Flux Isotope Reactor) which is a 100 Mw light water reactor with a Be reflector. It is located at the Oak Ridge National Laboratory where it has been used during the last 20 years as a source of neutrons for the purposes summarized in the introduction. Our model calculated a maximum thermal neutron flux in the reflector of $1.74 \times 10^{15}/cm^2 sec$ and a core life of 27 days; the real core life is 23 days. We found the results satisfactory after considering that our model did not include the effects of the beam tubes.

The details of a preliminary design of the new reactor for the proposed Center for Neutron Research (CNR) are summarized in Table II. The thermal flux in the reflector per unit power is $2.65 \times 10^{13}/cm^2 secMw$ a number to be compared with the corresponding parameter of HFIR, $1.74 \times 10^{13}/cm^2 secMw$. The CNR reactor would be then 52% more efficient than HFIR. The improvement does not imply large changes of either the fissile loading (9.7 Kg HFIR vs 13.05 Kg CNR) or the core life (23 days HFIR vs 16 days CNR). Despite the higher efficiency the generation of a thermal flux in excess of $5 \times 10^{15}/cm^2 sec$ requires a high power density (the CNR reactor would more than double the power density of HFIR) which causes unprecedented thermal-hydraulic conditions. Research in this direction is in progress and outside the scope of this paper.

IV. CONCLUSIONS

A detailed neutronic analysis was made to design a new high flux reactor to be operated in the proposed Center for Neutron Reseach at Oak Ridge. The multiple use of the reactor together with the requirement of a high thermal flux and a minimum core life of two weeks constitute a major challenge for designers. The selection of a very undermoderated core and a very low absorbing reflector improves the efficiency relative to current designs allowing unprecedented fluxes to be made available at a reasonable power requirement.

REFERENCES

1. Physics Today, January 1985.

2. Lamarsh J. R. Introduction to Nuclear Reactor Theory, p. 206, Addison-Wesley, 1966.

3. Vondy D. R., Fowler T. B., and Cunningham G. W. The Bold-Venture Computation System for Nuclear Reactor Core Analysis, Version III, ORNL-5711 (1981).

4. Cheverton R.D., and Sims T. M. HFIR Core Nuclear Design, ORNL-4621 (1971).

Table I. Power (P) to produce a $10^{16}/cm^2 sec$ thermal flux[a]

Case	V (liters)[b]	L_1 (cm)	D_1 (cm)	P (Mw)	ρ (Mw/1)[c]	F[d]
1	0.	-	-	50.	∞	1.000
2	40.	3.15	0.849	193.	4.82	0.906
3	60.	5.11	1.026	214.	3.57	0.872
4	90.	6.48	1.091	251.	2.79	0.830
5	120.	7.46	1.123	285.	2.37	0.795

[a] Spherical designs, D_2O moderator/reflector (thickness d= 100 cm), 42% metal fraction, only critical loading (zero core life).

[b] Volume of the core.

[c] Power density.

[d] Fraction of fission neutrons moderated in the reflector.

Table II. Preliminary CNR reactor parameters

Core:	Fuel 93% enriched U, Al cladding, volume 41.8 liter, metal fraction 0.31, fuel loading 13.05 Kg ^{235}U, moderator D_2O, burnable poison 6g ^{10}B
Reflector:	D_2O, thickness 100 cm radially, 73 cm axially
Target region:	at the center, 2.7 liters
Control rods:	in radial reflector
Flux:	maximum in radial reflector $5.31 \times 10^{15}/(cm^2 sec)$ (beginning of cycle), $6.25 \times 10^{15}/(cm^2 sec)$ (EOC) at 200 Mw
Core life:	16 days at 200 Mw