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OAK RIDGE, TENNESSEE
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CENTRAL FILES NUMBER

CF-57-7-23

External Transmittal Authorized

COPY NO. 126

DATE: July 10, 1957

SUBJECT: Nuclear Computations for HRE-3 Design:
Equilibrium Results

TO: Distribution

FROM: M. W. Rosenthal and T. B. Fowler

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ABSTRACT

Various nuclear characteristics of two-region spherical homogeneous reactors have been computed in order to provide information for the design of ERE-3. Equilibrium isotope concentrations were established using an ORACLE code, and a two-group model was used to obtain critical concentrations and flux distributions.

Breeding ratio is plotted as a function of reactor size, blanket thorium concentration, and other design and operating parameters, and the time required for a demonstration of breeding is discussed. Tables of results, including neutron balances, are given for selected reactors. A number of relations are presented for estimating the effects of fission products, copper, corrosion products, H_2O , and the core tank on breeding ratio.

INTRODUCTION

Nuclear computations have been performed to determine the effects of various design parameters on the characteristics of spherical reactors in the size range likely for HRE-3. The ORACLE code employed ("Thorobred-I"⁽⁵⁾) computes the concentrations of isotopes which would exist at equilibrium, and uses a two-group model to estimate the critical concentration and flux distribution.

In the computations the following conditions were specified:

- a. The reactor system is at equilibrium with regard to isotope concentrations.
- b. Hydroclone separation of poisons from the core system is employed in addition to Thorex processing.
- c. The core Thorex cycle time is a dependent function of the specified total poison fraction, the blanket cycle time a function of the blanket U^{233} concentration.
- d. The poison fraction from Samarium is 0.8%, that from Xenon 1%.
- e. The external core system has 1.0 liter of volume for every 20 kw of core power; the blanket external system has 1.0 liter for every 14 kw of blanket power.
- f. The average core and blanket temperature are 280°C.

Because of the importance of demonstrating a positive breeding gain in the HRE-3, the effects of design and operating conditions on breeding gain have been examined in detail. Core-wall power densities, neutron balances, inventories, and other results are also presented and discussed.

It should be noted that the two-group model used for the nuclear computations is a gross simplification of the neutron behavior. While useful for survey calculations, it possibly does not yield very accurate critical concentrations and breeding ratios, particularly for cores as small as 4 ft in diameter. Multigroup computations are being performed to determine the reliability of the two-group values.

An additional simplification has been the assumption of equilibrium concentrations of isotopes and poisons. Actually a reactor would be started with little poison present, and the isotopic composition of uranium and protactinium would be appreciably different from that at equilibrium. Many years of operation are required before equilibrium conditions are approached. As a result, the breeding gains obtained for steady state concentrations are lower than would occur in a reactor started "clean," particularly during the first several years. An ORACLE code which will estimate reactor behavior under transient conditions is in preparation. When completed, it will be used to predict rates of U^{233} production and average breeding gains when reactor conditions are changing with time.

RESULTS

Values of breeding ratio and blanket power are plotted in Fig. 1 - 3. In Tables 1 and 2 more extensive results, including neutron balances, are given for selected reactors. The neutron balances are normalized to 100 absorptions in U^{233} and U^{235} ; therefore, the numerical values represent approximately the percentage effect of the various items on the breeding ratio. (Note, however, that the effect of Pa^{233} losses on breeding ratio would be obtained by doubling the values given.)

Breeding Ratio

The breeding ratio is a quantitative measure of the efficiency of a reactor as a producer of fuel and is defined as follows:

$$BR = \frac{\text{Rate of fuel production}}{\text{Rate of fuel destruction}} \quad (1)$$

A reactor having a breeding ratio greater than 1.0 produces more fuel than it consumes.

Reactor Size -- The dependence of breeding ratio on pressure vessel size, core diameter, and blanket thorium concentration is shown in Fig. 1.

Although the isotope concentrations were obtained for a total power (heat) of 61.5 Mw, the results would differ little at other power levels.

Fission-Product Poisons -- The effect of total poison fraction (ratio of absorption cross-section of poisons to fission cross-section of fuel) on breeding ratio is shown in Fig. 2. Since straight lines are obtained, the change in breeding ratio with change in poison fraction can be estimated by equation (2).

$$\Delta BR = -0.75 \Delta f_{pc} \quad (2)$$

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TABLE 1

CHARACTERISTICS OF SELECTED REACTORS AT EQUILIBRIUM

Core diameter, ft	4	4	4	4	5
Pressure vessel I.D., ft	8	8	9	9	9
Blanket thorium, g/liter	500	1000	500	1000	1000
Blanket U-233, g/kg Th	3.0	3.0	3.0	3.0	3.0
Core poison fraction	0.07	0.07	0.07	0.07	0.07
Concentration of U-233, g/liter	3.68	4.04	3.63	4.02	2.19
Concentration of U-235, g/liter	0.41	0.39	0.37	0.37	0.21
Breeding ratio	1.041	1.094	1.086	1.123	1.089
Core-wall power density (inside), kw/liter	27	23	27	23	10
Core-wall power density (outside), kw/liter	9	14	9	14	11
Core cycle time, days	833	901	817	893	616
Blanket cycle time, days	220	371	288	504	486
Core inventory of U-233 and U-235, kg	14.4	15.7	14.0	15.5	10.6
Blanket inventory of U-233 and U-235, kg	11.0	21.9	15.9	31.6	28.9
Net U-233 and U-235 production, g/day	2.5	5.8	5.3	7.5	5.4
Generation time, days	220	230	210	220	160
Doubling time, days	5760	2710	2640	2050	1950
Core system volume, liters	3520	3540	3500	3530	4420
Blanket system volume, liters	7320	7290	10580	10520	9610
Mean thermal flux in core system	1.3×10^{14}	1.3×10^{14}	1.4×10^{14}	1.3×10^{14}	1.8×10^{14}
Core power, Mw(heat)	51.5	51.9	51.0	51.7	51.2
Blanket power, Mw(heat)	10.0	9.6	10.5	9.9	10.3
Absorptions and leakages per 100 absorptions in fuel					
Absorptions in core by:					
U-233	74.7	76.4	74.6	76.4	75.3
U-234	9.2	8.2	8.4	7.8	8.2
U-235	9.1	8.1	8.3	7.6	8.0
U-236	0.7	0.4	0.4	0.3	0.4
Poisons	5.2	5.3	5.2	5.3	5.2
Heavy water	0.9	0.8	0.9	0.8	1.6
Absorptions in core tank	2.1	1.6	2.1	1.6	2.4
Absorptions in blanket by:					
U-233	16.2	15.5	17.1	16.0	16.6
U-234, U-235, U-236	0.1	0.1	0.1	0.1	0.1
Th	95.8	101.7	100.9	104.8	101.0
Pa-233	0.9	0.5	0.7	0.4	0.4
Poisons	0.4	0.4	0.4	0.4	0.4
Heavy water	0.5	0.2	0.5	0.2	0.2
Fast leakage	4.3	3.6	2.2	1.6	3.0
Slow leakage	3.4	0.9	1.7	0.4	0.8

Blanket U^{233} Concentration --- In Fig. 3 the effect of blanket uranium concentration on breeding ratio is indicated. The effect is small for thick or heavily loaded blankets, becoming more important if the blanket is thin or lightly loaded.

Xenon Retention --- Achievement of a xenon poison fraction of 0.01 requires the removal of most of the xenon formed in the reactor or its iodine precursor before neutron capture occurs. If there is no fast cycle system for iodine or xenon removal, the poison fraction resulting from xenon¹ will be about 0.05. According to equation (2), the breeding ratio values in Fig. 1 would thus be reduced by 0.03 if all xenon were retained.

Core Tank Absorptions --- The thickness of the Zircaloy core tank was taken as 0.42 in. for the 5-ft core and 0.33 in. for the 4-ft vessel. With these thicknesses, as shown in Table 1, neutron captures in zirconium reduce the breeding ratio about 0.02 in the 4-ft core and 0.03 in the 5-ft core. If the core-tank thickness were altered, the losses would be changed proportionately.

Absorptions in Copper --- No allowance for neutron absorptions in the copper recombination catalyst was made in the computations. The poison fraction attributable to copper in various concentrations and the effects on breeding ratio (ABR) are given in the following table.

Table 2 Effect of Copper on Breeding Ratio

<u>Core Diameter</u>	<u>Copper Concentration</u>	<u>Poison Fraction</u>	<u>ABR</u>
4 ft	0.01 g-mol Cu/liter	0.004	-0.003
4	1.0 g Cu/liter	0.006	-0.005
5	0.01 g-mol Cu/liter	0.008	-0.006
5	1.0 g Cu/liter	0.013	-0.010

For other concentrations the poisoning effects are proportionate.

The copper concentration required for 100% recombination in a 4-ft core at 60 Mw has been estimated⁽⁴⁾ at 0.018 g-mol/liter. The loss of neutrons to copper at this concentration and core size would reduce the breeding ratio by 0.005.

H₂O Contamination --- Any H₂O contained in the heavy-water moderator will act as a poison and reduce the breeding ratio. In the calculations which have been performed there was assumed to be 0.25% H₂O in the moderator. Neutron captures in moderator in a 4-ft core (see Table 1) were found to be about 0.009 per absorption in fuel, of which about 60%, or 0.005, were in H₂. Thus, 0.25% H₂O, which is 9 liters in a 3600-liter system, reduced the breeding ratio by 0.005. Other values are given in the table which follows.

Table 3. Effect of H₂O on Breeding Ratio

<u>Core Diameter</u>	<u>H₂O</u>	<u>ABR</u>
4	1.0%	- 0.02
4	18 liters	- 0.01
5	1.0%	- 0.04
5	11 liters	- 0.01

Other amounts of H₂O would have effects in proportion to the concentrations.

A specified volume of H₂O added to the blanket has much less effect on breeding ratio than the same amount in the core. This is a result both of the lower flux in the blanket region and the larger volume of the system.

Corrosion Products --- Assuming the surface area of stainless steel in the core high-pressure system to be 6000 sq ft, corrosion to an average depth of 0.001 in. would remove 250 lb of metal. If this were distributed uniformly

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throughout the fuel solution, the poison fraction resulting from it would be about 0.18 in a 4-ft core and 0.36 in a 5-ft core.

However, iron and chromium are precipitated⁽²⁾ and would be removed by hydroclones. If the hydroclones were operated on a fast cycle time (say several days), neutron capture in iron and chromium would be unimportant. Nevertheless the absorption cross sections of the nickel and manganese (which probably remain in solution) amount to about 15% and 9%, respectively, of the total absorption cross section of 347 stainless steel. Thus, from 0.001 in. of corrosion, the nickel and manganese would yield a poison fraction of about 0.04 (ABR = 0.03) in a 4-ft core and 0.08 (ABR = 0.06) in a 5-ft core.

The actual value of the poison fraction from corrosion products would, of course, depend on the corrosion rate and the chemical processing rate. If the corrosion products are assumed to change to isotopes of the same cross section on neutron capture, the following relations are obtained:

$$f_p = 0.04 \times R \times (T_c/365) \quad (\text{Core ID} = 4 \text{ ft}) \quad (3)$$

$$f_p = 0.08 \times R \times (T_c/365) \quad (\text{Core ID} = 5 \text{ ft}) \quad (4)$$

In equations (3) and (4), f_p is the equilibrium core poison fraction from corrosion products, R the mean corrosion rate in mils/yr, and T_c the core cycle time in days. These equations are based on the conditions postulated above, namely: 6000 sq ft of surface; rapid removal of iron and chromium, but not nickel and manganese. The assumption regarding the unchanged cross sections of fission products is probably conservative, since the average cross section appears to become smaller.

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An additional point of concern resulting from corrosion of stainless steel is the adverse effect of high corrosion-product concentrations on the stability of fuel solution. The concentration of nickel from 0.001 in. of corrosion would be 0.052 g-mol/liter, that of manganese would be 0.011 g-mol/liter. Unless adjustments were made to the acid concentration, there would probably be formation of a second phase before the above concentration of nickel was attained.

Transfer of zirconium from the core vessel into the fuel solution by corrosion would have negligible effect on the breeding ratio. While some zirconium would be moved from the wall to a more important region in the core, this would be more than offset by the several-fold dilution of the external circulating system.

Generation and Doubling Times

The gross rates of fuel generation in the blanket can be estimated using the equations below (based on 185 Mev/fission):

$$G = 1.25 \times P_t \times BR \text{ (U}^{233} \text{ fuel)} \quad (5)$$

$$G = 1.34 \times P_t \times BR \text{ (U}^{235} \text{ fuel)} \quad (6)$$

The generation rate, G , is in grams of U^{233} per day, P_t is total reactor power in Mw, and BR is the breeding ratio. The net generation rate in the blanket would be obtained by subtracting the rate at which uranium is consumed in the blanket. Computation of the net production rate in the system (given in Table 1) requires subtraction of the fuel consumption in the core as well. For convenience, the generation time of a two-region reactor is defined here as the time required to accumulate an amount of fissionable material in the blanket equal to the core system inventory.

This does not at all represent a net production of fuel, since approximately one core inventory is consumed in the process. The generation time in the HRE-3 with a 4-ft core would be about 220 days (see Table 1) and for a 5-ft core would be about 150 days.

At the beginning of a breeding demonstration it would be desirable to have about two core inventories of U^{233} on hand: one for the critical loading, a second as feed for the period lasting until blanket material can be generated, chemically processed (requiring perhaps 100 days), and fed to the core. When this much U^{233} is available, the reactor can be operated on the U^{233} - Th cycle without an outside source of fuel. If U^{233} is not available at the time of reactor startup, it will be necessary to consume U^{235} in the core while generating U^{233} in the blanket. Since η^{25} is only 2.08 compared to 2.25 for η^{23} , the breeding ratio will be lower and the generation time will be somewhat longer, perhaps 20%. Over 500 days would be required to produce two core inventories of fuel using U^{235} in a 4-ft core, over 350 days with a 5-ft core.

The time required for a breeding demonstration is dependent on the concentration of U^{233} to be attained in the blanket. The times given below are estimates of the periods required for the U^{233} concentration in the blanket to reach 1 g/kg Th if U^{233} is used in the core.

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Table 4. Time Required to Attain 1.0 g U²³³/kg Th

Core ID, ft	Pressure Vessel I.D. ft	Blanket Th, g/liter	Production Time, days
4	8	500	60
4	9	500	80
4	10	500	110
5	9	500	70
5	10	500	100
4	8	1000	110
4	9	1000	150
4	10	1000	210
5	9	1000	140
5	10	1000	200

The doubling time, as defined here and as given in Table 1, is the period required for the production of a net excess of fuel equal to the core inventory. This time is dependent on the breeding gain (BR-1.0) rather than the breeding ratio, and hence varies considerably with the reactor design, ranging from 1400 to 5800 days for the reactors considered.

Core-Wall Power Density and Blanket Power

Some values of the power density (heat generation rate) in the fuel solution at the inside and outside surfaces of the core vessel are given in Table 1. For a 4-ft core vessel with 3.0 g U²³³/kg Th in the blanket, the inside core-wall power density is between 22 and 27 kw/liter; for a 5-ft vessel the range is between 9 and 12 kw/liter. The core-wall power density would be about 15% higher if there were no uranium in the blanket (with the same total power).

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The fraction of total fission energy which is released in the blanket is plotted in Fig. 3 as a function of blanket U^{233} concentration. Although not included, at the lower U^{233} concentrations the gamma heat removed by the blanket may be an appreciable fraction of the total blanket power. (3)

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