REACTOR PHYSICS AND FUEL-CYCLE ANALYSES

REACTORS

A. M. PERRY and H. F. BAUMAN
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

KEYWORDS: *molten-salt reactors, fuels, economics, operation, breeding, thorium-232, uranium-233, performance, MSBR, fuel cycle, cost, breeding ratio*

Received August 4, 1969
Revised October 9, 1969

As presently conceived at Oak Ridge National Laboratory and described in this issue, the single-fluid Molten-Salt Breeder Reactor, operating on the ^{232}Th-^{233}U fuel cycle and based on a reference design, has a breeding ratio of ~1.06, specific fissile inventory of 1.5 kg/MW(e), a fuel doubling time of ~20 years, and fuel cycle costs of ~0.7 mill/kWh(e). Start-up may be accomplished with either enriched uranium or plutonium, with little effect on fuel cost; the breeding ratio, averaged over reactor life, is reduced 0.01 to 0.02 relative to the equilibrium cycle.

Operated as a converter, with limited chemical processing, the reactor may have a conversion ratio in the range 0.8 to 0.9 with fuel cycle costs of 0.7 to 0.9 mill/kWh(e).

INTRODUCTION

One of the most important aspects of the Molten-Salt Reactor (MSR) concept is that it is well suited for breeding with low fuel-cycle costs, and it does so in a thermal reactor operating on the ^{232}Th-^{233}U fuel cycle. This is true not primarily because of any unique nuclear characteristics, for the reactor is similar to other thermal reactors in terms of attainable fuel-moderator ratios, the unavoidable presence of certain parasitic neutron absorbers, and reliance on a fertile blanket to reduce neutron losses by leakage to an acceptably low level for breeding. Indeed, the concept might be thought to have some *a priori* disadvantage, because a substantial fraction of the fissile material is invested in the heat transfer circuit and elsewhere outside the reactor core. The peculiar suitability of the molten-salt reactor for economical thermal breeding stems rather from the practical possibility of continuous removal of fission-product wastes and ^{233}Pa, and virtually arbitrary additions of uranium or thorium, without otherwise disturbing the fuel. This fundamental aspect of the molten-salt reactor, details of which are discussed in other papers of this series, has a profound effect on the relationship between neutron economy and fuel-cycle cost. The coincidence of good neutron economy with low fuel-cycle cost which characterizes the molten-salt reactor appears to be unique among thermal reactors and will be described more fully in this paper.

GENERAL NUCLEAR CHARACTERISTICS

The LiF/BeF_2 carrier salt used in the MSR concept is not by itself a very good moderator. Its moderating power is about half to two-thirds that of graphite (the exact value depending on the proportions of Li and Be in the salt), while its macroscopic absorption cross section is an order of magnitude greater than that of graphite, even with the feed lithium enriched to 99.995% in the ^{7}Li isotope. (With this composition, <10% of the neutron absorptions in the salt occur in ^{6}Li; nearly half are in fluorine, and about a third in ^{7}Li.) It is evident, therefore, that an additional moderator is needed, and graphite is selected for this purpose because of its compatibility with the salt.

There is only a weak connection between the fissile fuel concentration in the carrier salt and the heat transfer characteristics of the salt (arising primarily from the influence of the thorium concentration on the physical properties of the salt), and as a consequence one has considerable latitude in selecting the uranium (and thorium) concentrations in the salt. Because the carrier salt itself constitutes a significant neutron poison, the fuel concentration in the salt must not be set at too low a level, but must be high enough for the fuel to compete favorably (for neutrons) with the lithium and the fluorine in the salt. On the other hand, it must not be too high, lest the inventory of fuel outside the reactor core become excessive. The optimum fuel concentration, typically ~0.2 mole% of UF_4 in the salt, or ~1 kg of uranium per cubic foot of salt, is interrelated with the neutron spectrum in the reactor, which is a function of the relative proportions of fuel salt and graphite moderator in the core. Too large a proportion of salt leads to an excessive fuel inventory and to a poorly thermalized neutron spectrum, with a reduced neutron yield, η; too large a proportion of graphite leads to excessive neutron-absorption losses in the graphite. An optimum salt volume fraction is typically found to be ~13 to 15%.

The proper balance of the above factors does, of course, depend in part on the power density in the reactor core, which may be selected almost independently of the power density in the remaining parts of the primary salt circuit. The maximum power density in the core is limited by fast neutron damage to the graphite moderator, while the removal power density in the external power recovery circuit is limited primarily by heat transfer and pressure-drop considerations and by requirements for pipe flexibility in the piping runs between the reactor vessel and the heat exchangers.

The necessity for maintaining a sufficiently high fuel concentration to suppress neutron losses in the carrier salt and in the moderator, together with the requirement for appreciable core size simply to generate the requisite amount of power, leads to the conclusion that thorium must be present in the core, not merely in a surrounding blanket. However, the question of how the thorium is to be incorporated in the core is crucial to the MSBR concept. One quickly recognizes several distinct possibilities, some much more desirable in principle than others, but full of implications with respect to reactor design and chemical processing.

We have previously given serious consideration to a two-fluid reactor in which the fissile and fertile materials are carried in separate salt streams, the bred uranium being continuously stripped from the fertile stream by the fluoride volatility process. Blanket regions contain only the fertile salt, while the core contains both fissile and fertile streams; these streams must be kept separate by a material with a low-neutron cross section, that is, by the graphite moderator itself. This approach appears to yield the best nuclear performance, owing primarily to a combination of maximum blanket effectiveness and minimum fuel inventory. It also exhibits attractive safety characteristics because expansion of the fuel salt, upon heating, removes fissile material from the core while leaving the thorium concentration unchanged. The concept does, however, involve important questions regarding the reliability of the graphite "plumbing" in the core, the adequate proof of which may require a good deal of time and testing.

The present approach employs a single salt stream which contains both the fissile and the fertile materials. This concept represents a modest extrapolation of the technology already demonstrated in the MSRE. A central feature of the concept is the manner in which the single salt composition can be made to function adequately both in the core and in the blanket (or outer core) regions. This is done by the simple expedient of altering the salt volume fraction, making it considerably larger in the blanket than in the core. This undermoderation results in enhanced resonance capture of neutrons by thorium in the outer region, gives rise to a negative material buckling in the outer region, and should in principle cause a fairly rapid decrease in power density in the blanket as a function of distance from the core boundary. In practice, the distinction between the core and blanket regions is not as clear cut as this argument may suggest, but the idea works reasonably well. Figure 1 illustrates the power density distribution for our present reference design based on the single-fluid concept. The enhancement of resonance neutron capture in the blanket (or outer core) region is indicated by the ratio of neutron absorptions in ^{232}Th to those in ^{233}U; this ratio is about 1.0 in the core, and 1.3 in the blanket. The salt annulus, which is required to allow the periodic replacement of the moderator, functions as a part of the outer core region.

The principal shortcoming of the single-fluid concept, of course, is the substantial investment of fissile material in the blanket region. This results in a rather different compromise between breeding gain and specific inventory than in the two-fluid concept, leading both to reduced effectiveness of the blanket region and to an appreciable increase in fuel inventory. Fortunately, this feature of the single-fluid reactor is partly offset by a reduction in neutron captures in the carrier salt, owing to the fact that a single carrier salt contains both fissile and fertile materials.

The preceding qualitative discussion is intended to provide a general understanding of the interplay of factors affecting the selection of MSBR design parameters. These factors are of course quite numerous. They include core size, radial and axial blanket thickness, reflector thickness, salt volume fractions in the core and blanket regions, thorium and uranium concentrations,

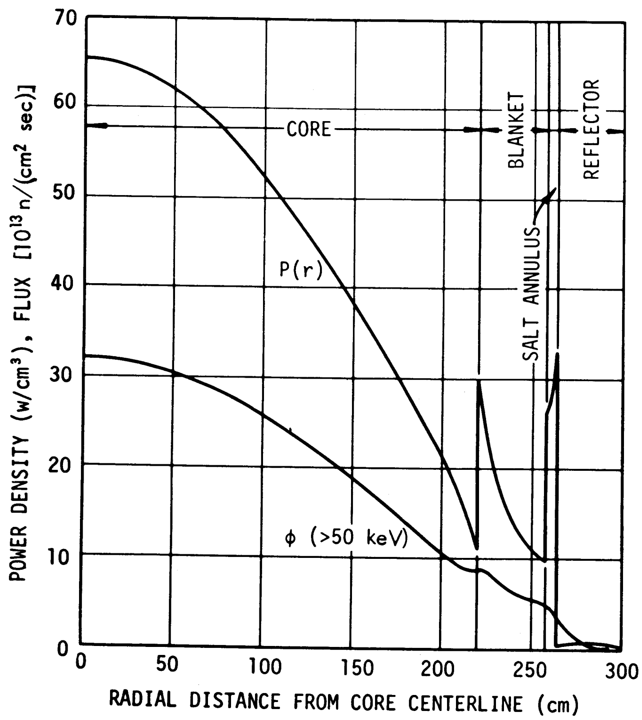

Fig. 1. Radial power density and fast flux distributions—single-fluid MSBR.

chemical processing rates, and reactor power level. Because the interaction of all these factors is rather complex, and because of the need to identify optimum values of the design variables rather closely, we have found it convenient to make use of a comprehensive, automatic reactor optimization procedure for arriving at that combination of design parameters that will produce, in some sense, the best attainable performance. The Reactor Optimization and Design code (ROD) is based on a gradient projection method for locating the extreme value of a specified figure of merit, which may be any desired function of the breeding ratio, the specific fuel inventory, various elements of the fuel cycle and capital costs, or any other factors important to the designer. The computational procedure comprises multigroup (synthetic), two-dimensional diffusion-theory calculations of the neutron flux, an equilibrium fuel-cycle calculation which determines the critical fuel concentration and nuclide composition consistent with processing rates and other variables, and the gradient projection calculation for moving the cluster of independent variables in the direction that most rapidly improves the figure of merit. The optimization may be constrained by limiting the allowed range of the independent variables, or by selecting in advance the desired value (or a limiting value) of certain derived quantities, such as the maximum power density.

The figure of merit used here in determining reactor design specifications is related to the capability of a reactor type to conserve fuel supply in an expanding nuclear economy. For the special case of a linear increase in power generation, the total amount of natural uranium that must be mined up to the point when the system becomes self-sufficient (i.e., independent of any external supply of fissionable material) is proportional to the product of the doubling time and the specific fuel inventory. We have chosen to optimize our MSBR design primarily on the basis of a quantity which we call the fuel "conservation coefficient," defined as the breeding gain times the square of the specific power, which is equivalent to the inverse of the product of the doubling time and the fuel specific inventory. Therefore, a maximum value of the conservation coefficient is sought in the optimization procedure.

EQUILIBRIUM FUEL-CYCLE RESULTS

The result of a reactor optimization calculation is a set of specifications for the optimum reactor configuration, subject to any imposed constraints, together with a complete description of its equilibrium fuel cycle. This description includes the multigroup neutron flux distributions, the resulting power distribution, and the consistent set of concentrations of all nuclides present in the reactor. We have imposed constraints on maximum power density (i.e., minimum graphite life), on overall reactor vessel dimensions, and on chemical processing rates which we believe will result in near-minimum power cost. Although we lack specific information as to the cost of chemical processing as a function of fuel processing rate for the liquid-metal extraction process, it appears that processing equipment sizes and operating costs will be comparable with those for the fluoride-volatility/uranium-distillation process considered for the two-fluid reactor. We have therefore fixed the processing rates, listed in Table I, at values found to be essentially optimum in studies of the two-fluid reactor, with minor adjustments appropriate to the extraction process. While subsequent improvements in processing cost estimates may suggest some change in optimum processing rate and some change in fuel cost estimates, we do not expect that these will result in any major revision in performance estimates for the reactor.

The reference reactor configuration which results from these and other (engineering) considerations is described by Bettis.[1] A summary of its nuclear design characteristics is given in Table I.

TABLE I

Characteristics of the One-Fluid MSBR Reference Design

A. Description	
Identification	CC93
Power, MW(e)	1000
MW(th)	2250
Plant factor	0.8
Dimensions, ft	
Core zone 1	
Height	13.0
Diameter	14.4
Region thicknesses	
Axial: Core zone 2	0.75
Plenum	0.25
Reflector	2.0
Radial: Core zone 2	1.25
Annulus	0.167
Reflector	2.5
Salt fractions	
Core zone 1	0.132
Core zone 2	0.37
Plena	0.85
Annulus	1.0
Reflector	0.01
Salt composition, mole%	
UF_4	0.228
ThF_4	12
BeF_2	16
LiF	72
Processing cycle times for removal of poisons[a]	
1. Kr and Xe; sec	20
2. Se, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Sb, Te, Zr; sec	20
3. Pa; Cd, In, Sn; days	3
4. Y, La, Ce, Pr, Nd, Pm, Sm, Eu, Gd; days	50
5. Sr, Rb, Cs, Ba; year	5
6. Br, I; days	5

B. Performance	
Conservation coefficient, $[MW(th)/kg]^2$	14.3
Breeding ratio	1.062
Yield, % per annum	3.18
Inventory, fissile, kg	1478
Specific power, MW(th)/kg	1.52
Doubling time, system, year	22
Peak damage flux, E >50 keV, $n/(cm^2\ sec)$	
Core zone 1	3.2×10^{14}
Reflector	4.2×10^{13}
Vessel	3.7×10^{11}
Power density, W/cm^3	
Average	22.2
Peak	65.2
Ratio	2.94
Fission power fractions by zone	
Core zone 1	0.765
Core zone 2	0.167
Annulus and plena	0.056
Reflector	0.012

[a]According to our present flow sheet, Zr, Cd, In, and Sn will be removed on a 200-day cycle, and Br and I on a 50-day cycle. The additional poisoning, however, is negligible.

A neutron balance for this case is given in Table II, in which the normalization is to one neutron absorbed in ^{233}U plus ^{235}U.

Uncertainties in Neutron Cross Sections

We have estimated the effect of uncertainties in neutron cross sections on the calculated performance of the MSBR. By far the most important effect is the uncertainty in the average value of η of ^{233}U in the MSBR spectrum, which leads to an uncertainty in the breeding ratio of ±0.012. Uncertainties in the cross sections of other important MSR nuclides (such as F) make a relatively small contribution to the overall uncertainty in the breeding ratio, which is estimated to be ±0.016. A detailed discussion of cross-section uncertainties is given in a report by Perry.[2]

Equilibrium Fuel-Cycle Costs

As stated before, the molten-salt breeder reactor exhibits unusually low fuel-cycle costs in combination with good breeding performance. This results primarily from the low specific fuel inventory and from a small but non-negligible excess production of fuel, which results from the ability to process the fuel rapidly at what appears to be a very low unit cost.

The inventory of fissile material in the reactor and chemical processing plant amounts to some 1480 kg, including ~100 kg each of ^{235}U and ^{233}Pa; when valued at \$13.00/g for ^{233}U and ^{233}Pa and \$11.20/g for ^{235}U, this material is worth \$19 million. With an effective annual inventory charge rate of 10%/year and a 0.8 plant factor, the fuel inventory thus contributes 0.27 mill/kWh(e) to the