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Summary

This report is a compilation of the results of reactor physics calculations to date for the currently proposed MSRE core design. The core was assumed to consist of a homogeneous mixture of fuel salt and graphite, with 22.5 per cent of the core volume occupied by fuel; the salt composition was the currently proposed mixture of 70 mole per cent LiF, 23 mole per cent BeF_2, 5 mole per cent ZrF_4, 1 mole per cent ThF_4, and UF_4 as required for criticality. The calculated critical mole per cent, assuming 93.5 per cent U-235, is 0.2 mole per cent UF_4; the associated inventory of U-235 in the circulating system is 45 kilograms. Mean core thermal flux is estimated to be 2.9×10^{13} n/cm^2 sec with an associated mean power density of 3.9 watts/cm^3 for 10 megawatts total reactor power.

MSRE PRELIMINARY PHYSICS REPORT

C. W. Nestor, Jr.

Introduction

The purposes of this report are to assemble the results of the reactor physics calculations which have been done concerning the currently proposed MSRE core design, and to point out the areas in which further work needs to be done. Estimates have been made of the reactor characteristics using the core model and calculation methods discussed in Reactor Model and Calculation Methods; these results are presented in Table 1 and discussed in Results. Consideration is given to the problems of fission product buildup, fuel salt and Xe-135 retention by the core graphite, and distortion of the core graphite under irradiation in Long-term Reactor Behavior. It should be emphasized that in some cases these results depend upon very scanty experimental data buttressed by many assumptions and that much more work remains to be done in this particular area.

Reactor Model and Calculation Methods

For the criticality calculations the reactor was assumed to be a bare right circular cylinder 27.7 inches in radius and 63 inches high; a radial extrapolation distance of 1 inch was added to simulate the effect of the fuel annulus and INOR-8 vessel, and an axial extrapolation distance of 3.5 inches was added to both ends to simulate the fuel salt contained in upper and lower heads of the vessel. The IBM-704 multigroup one-dimensional diffusion theory program GNU-II[1] was used for the calculations with the 34-group cross section library prepared for use in the thorium reactor evaluation program.[2] The core was assumed to be a homogeneous mixture of 77.5 volume per cent graphite (density 1.90 gm/cm^3) and 22.5 volume per cent fuel salt, using the currently proposed mixture[3] of 70 mole per cent LiF (99.997% Li^7), 23 mole per cent BeF_2, 5 mole per cent ZrF_4, 1 mole per cent ThF_4 and ~ 1 mole per cent UF_4 (as required for criticality). The external circulating system volume was assumed to be 40 ft^3, which gave a ratio of total circulating system fuel volume to core fuel volume of 3.0. The temperature and concentration coefficients of reactivity were estimated from the output of the criticality search section of the GNU program, as previously described.[4]

Two-dimensional two-group flux calculations were done using the IBM-7090 program Equipoise-II[5] to obtain estimates of the power generated in the upper and lower heads of the vessel and in the fuel annulus surrounding the core. This program was also used in the estimation of the effects of graphite distortion on reactivity (see Long-term Reactor Behavior). Two-group constants were obtained from the output of the GNU program.

Results

The principal results are tabulated in Table 1.

Table 1. Reactor Physics Data for the MSRE

Shape	Right circular cylinder	
Core size	Radius 27.7 inches, height 63 inches, volume 88 ft^3	
Fuel volume fraction	.225	
External fuel volume	40 ft^3	
Total fuel volume/core fuel volume	3.02	
Temperature	1200°F	
Power	10 megawatts	
Graphite density	1.90 gm/cm^3	
Fuel salt composition:	component	mole percent
	LiF	70.6
	BeF_2	23.2
	ZrF_4	5.0
	ThF_4	1.0
(Clean critical)	UF_4	0.21 (93.5% U^{235})
Circulating system U^{235} inventory*		45 kg
Mean core thermal flux		2.9×10^{13} n/cm^2 sec
Peak core thermal flux		7.4×10^{13} n/cm^2 sec
Mean core power density		3.9 $watts/cm^3$
Peak core power density		10 $watts/cm^3$
Specific power		40 kw/kg of U + Th
Temperature coefficients of reactivity:		
fuel salt		$-3 \times 10^{-5}/°F$
graphite		$-6 \times 10^{-5}/°F$
U^{235} concentration coefficient, $\frac{\delta k/k}{\delta C_{25}/C_{25}}$		0.25
Equilibrium Xe^{135} $\delta k/k$ (see Long-term Reactor Behavior)		1.3%
Equilibrium Sm $\delta k/k$		0.7%
Neutron lifetime		3×10^{-4} sec
Per cent of fissions due to thermal neutrons		87
Fraction of power generated in core		0.96

*Addition of 2% poison raises critical mass by about 8%.

Long-Term Reactor Behavior

In the currently proposed MSRE core the fuel salt is in contact with the graphite moderator and some penetration of the graphite by gaseous fission products and by fuel salt will certainly occur. It is, however, extremely unclear at this time what the amounts of these penetrations will be, since there is no experimental data concerning the behavior of fuel salt, and fission products in contact with the proposed MSRE graphite. In addition, gaseous fission products will be stripped from the salt in the pump bowl by a helium sparge when the reactor is operating at power. Any calculation dealing with the effects of fuel and fission product retention is therefore based on assumptions of unknown reliability and should be regarded only as an estimate of possible behavior. Using a particular set of assumptions concerning fuel salt and fission product behavior, efficiency of stripping in the pump bowl and graphite properties, Spiewak[6] has calculated an equilibrium Xe-135 poison fraction (ratio of Xe-135 atoms destroyed by neutron absorption to fissions) of .0184; this represents a reactivity change ($\delta k/k$) of 1.3% and this value is quoted in Table 1. If all the fuel and Xe-135 were fixed in the core, the associated reactivity would be 4%; there is a relatively wide range of values which may result from apparently equally reasonable assumptions.

Under long-term irradiation it is known that graphite will change its dimensions. Since no irradiation experiments have been done with the proposed MSRE graphite, the situation with regard to long-term reactivity changes is unclear. Using the results of calculations of graphite distortion by Kinyon,[7] it is estimated that the combined effects of graphite distortion and fission product buildup will amount to a reactivity decrease of 3.8% in one full power year's operation. These calculations were based on a single short-term experiment on a similar grade of graphite, not exposed to fuel salt; this result should therefore be regarded as only an estimate of possible behavior.

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