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SUBJECT: Survey of Low Enrichment Molten-Salt Reactors

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FROM: H. G. MacPherson

Abstract

A rough survey of the nuclear characteristics of graphite-moderated molten-salt reactors utilizing an initial complement of low enrichment uranium fuel has been made. Reactors can be constructed with initial enrichments as low as 1.25% U-235; initial conversion ratios of as high as 0.8 can be obtained with enrichment of less than 2%. Highly enriched uranium would be added as make-up fuel, and such reactors could probably be operated for burnups as high as 60,000 Mwd/ton before buildup of fission products would make replacement of the fuel desirable. A typical circulating fuel reactor of this class might contain an initial inventory of 3600 tons of 1.8% enriched uranium, operated at 640 Mw (thermal), and generate a net of 260 Mw (electrical). The total fuel cycle cost would be approximately 1.3 mills/kwhr, of which 1.0 mill is burnup of enriched U-235.

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Errata for CF 58-10-60

Please make the following changes in your copy of CF 58-10-60:

Page 1 (Cover Sheet) Make change in line 10 of abstract

From: 3600 tons of 1.8%

To: 36 tons of 1.8%

Page 2 Make change in next to last line

From: 8.54×10^{20}

To: 8.54×10^{22}

Page 3 Make change in line 18

From: $\sqrt{\frac{S}{n}}$

To: $\sqrt{\frac{S}{M}}$

SURVEY OF LOW ENRICHMENT MOLTEN-SALT REACTORS

To survey the field of low enrichment graphite-moderated reactors a number of calculations have been made with the four-factor formula $k_\infty = \eta\epsilon pf$. The following fuel salt was considered:

	mole %
UF_4	20
Li^7F	70
BeF_2	10

This salt has a melting point of about 900°F. It is probably more corrosive than one mole % fuel, but is probably satisfactory for use with INOR-8. The atomic concentrations were as follows at 600 - 650°F, per L. A. Mann:

Li - 175.7×10^{20} atoms/cc
Be - 25.12×10^{20} atoms/cc
U - 50.22×10^{20} atoms/cc
F - 426.8×10^{20} atoms/cc

The slowing down power of the salt is 0.0228 cm^{-1}, composed of contribution as follows:

Fluorine	.0149
Li	.0051
Be	.0028

The concentration of UF_4 is 2.62 g/cc; that of uranium is 1.98 g/cc. Graphite was assumed to be of density 1.7, with 8.54×10^{20} atoms/cc, and a slowing down power of 0.0631 cm^{-1}. The graphite was assumed to have a σ_a = 0.0045 barns.

In the four-factor formula, η is taken for convenience to be that for U-235, while the thermal utilization factor f is defined as the proportion of thermal absorption in U-235. Thus, using cross sections derived from Fig. 2.3, page 2.20, of ORNL-2500, Part 2, η for U-235 is taken as $2.47 \times \frac{528}{650}$ or $\eta = 2.00$. Effective thermal fission cross sections and absorption cross sections are assumed to be 528 barns and 650 barns, respectively, corresponding to neutron temperatures of 600°C.

The fast fission factor was assumed to be 1.02, since the values calculated for graphite lattices vary from 1.02 to 1.04. This assumption is sufficiently accurate for survey purposes.

For the calculation of the resonance escape probability, the resonance integral was calculated from,

$$\sigma_r = 3.8 \left(\frac{\Sigma_s}{N_o} \right)^{0.42} + 24.7 \frac{S}{M}$$

In this formula, $\frac{\Sigma_s}{N_o}$ is the scattering cross section in barns per uranium atom within the fuel channel, and $\frac{S}{M}$ is the surface area of the fuel channel per gram of U-238 in the channel. The first term is the same as the resonance integral for an infinite medium of the fuel salt composition. This formula, involving the first power of $\frac{S}{M}$ instead of $\sqrt{\frac{S}{\eta}}$, was used because it is more logically extrapolated to the case of dilution of uranium with fuel salt. For uranium metal this reduces to the familiar $\sigma_r = 9.25 \times 24.7 \frac{S}{M}$. In the fuel considered here, $\frac{\Sigma_s}{N_o} = 43.3$, and $\sigma_r = 18.5 + 24.7 \frac{S}{M}$.

A series of reactors having tubular fuel channels on a square array is considered. The spacing of these channels was arbitrarily taken as 8 in. center to center, and the calculations are made for various volume fractions of fuel in the graphite from 0.05 to 0.25. The following table gives the fuel channel diameter, the value of $\frac{S}{M}$, the value of σ_r, and the value of the resonance escape probability p for the different volume fractions. p is calculated from $p = e^{-A}$ where A is given by,

$$A = \frac{N_u \sigma_r F}{(1 - F)(\xi\Sigma_s)_g + F(\xi\Sigma_s)_{salt}}$$

where N_u is the U-238 atom density in the fuel channel and F is the volume fraction of fuel in the core. Numerically,

$$A = \frac{50.22 \times 10^{20} \sigma_r F \times 10^{-24}}{0.0631 (-1-F) + 0.0228 F}$$

Table I

F	D	S/M	σ_r	p	$\eta\epsilon p$
.05	2.08 in.	.382	27.95 barns	0.891	1.82
.075	2.54	.313	26.25	0.848	1.73
.10	2.94	.271	25.20	0.807	1.65
.15	3.60	.221	23.97	0.733	1.495
.20	4.15	.192	23.25	0.653	1.332
.25	4.65	.171	22.73	0.584	1.191

The thermal utilization factor is given by,

$$f = \frac{F \epsilon N_u \hat{\sigma}_a^{\,u}}{F \epsilon N_u \hat{\sigma}_a^{\,u} + F N_u \sigma_a^{\,238} + F \Sigma_a^{\,salt} + (1 - F) \Sigma_a^{\,g} \gamma}$$

where e is the enrichment, $\hat{\sigma}_a^u$ is the effective absorption cross section for U-235, and γ is the thermal flux disadvantage factor. For this survey, γ is assumed to be 2.0, which seems a reasonable value based on ORNL-2500. This simplifying assumption was made to avoid calculating the flux distribution, and probably is the roughest approximation made in this survey. Numerically the above equation reduces to,

$$f = \frac{3.22\ F\ e}{3.22\ F\ e + 0.0137\ F + 0.0028\ F + 0.000768\ (1 - F)}$$

An easy computational form is,

$$\frac{1}{f} = 1 + \frac{0.0157 + \frac{0.00077}{F}}{3.22\ e}$$

To calculate the enrichment necessary to achieve a given k_∞, the following transformations are made,

$$k_\infty = \eta \epsilon p f$$

$$\frac{1}{f} = \frac{\eta \epsilon p}{k_\infty}$$

and

$$e = \frac{0.0157 + \frac{0.00077}{F}}{3.22 \left(\frac{\eta \epsilon p}{k_\infty} - 1 \right)}$$

The conversion factor is calculated as follows,

$$R_c = \frac{(1 - p) + p \times (\text{proportion of thermal absorptions in U-238})}{p \times (\text{proportion of thermal absorptions in U-235})}$$

From k_∞, B^2 is obtained from $B^2 = \frac{k_\infty - 1}{M^2}$

and M^2 is taken as,

$$M^2 = \frac{\tau_{graphite}}{1 - F} + L^2_{graphite} \times \text{(proportion of thermal absorptions occurring in graphite)}$$

The correction for $\tau_{graphite}$ is an approximate one, but not too important numerically. $\tau_{graphite}$ is taken as 324 cm^2 and $L^2_{graphite}$ is 2950 cm^2.

From B^2 the dimensions of a minimum volume cylindrical core were calculated using a reflector savings of 2 ft. This is less than used in ORNL-2500, but may still be optimistic because some fuel must be used in the reflector to cool it.

Table II gives the results of calculations for six values of the volume fraction of fuel in the core and for two values of k_∞. The volume of fuel in the core should be evaluated in terms of the external volume for a circulating fuel reactor, which is about 0.56 cu ft per thermal megawatt, or 360 cu ft for a 260 Mw (electrical) plant. It is evident that one must pay for conversion ratios above 0.8 with enrichments of over 2%.

The selection of an economically optimum reactor of this type requires a knowledge of the method of chemical reprocessing and its cost, and a way of calculating the effect of poisoning by buildup of fission products. The latter problem was looked at briefly, using as a basis the calculations of Blomeke and Todd (ORNL-2127), and assuming a buildup of Pu sub infinity as defined in the Brookhaven Geneva Paper 461. For the very long exposures it is probable that the one-group calculations can not give a good answer because of the large buildup of absorbing Pu isotopes. However, it might be possible to operate a reactor such as Case 8 of Table II for as

long as 30 years at 640 Mw (thermal) without a need for the U-235 inventory to increase by more than a factor of two, and with a breeding ratio averaging greater than 50%, without any reprocessing.

For purposes of estimating the fuel cycle cost, a life of 10 years without reprocessing was assumed. For this period, the U-235 concentration would probably not have to be increased over its initial value, and the breeding ratio should average at least 60%.

Applying the formula of ORNL-2500, but using the thermal cycle of the reference design molten-salt reactor (CF-58-5-3), the chemical reprocessing charges and fuel inventory charges are 0.034 mill/kwhr and 0.14 mill/kwhr, respectively. Ten-year depreciation and capital charges on the base salt amount to 0.093 mill/kwhr. Burnup of U-235 would be $\sim$1.0 mill/kwhr at a conversion ratio of 0.6. Thus, total fuel cycle costs would be approximately 1.3 mills/kwhr.

This reactor was based on Case 8 of Table II, using a total fuel volume of 600 ft^3 and an inventory of 36 tons of uranium of 1.8% enrichment. The ten-year reprocessing cycle represents a fuel life of approximately 60,000 Mwd/ton.

More accurate calculations are needed to confirm the above conclusions.

Table II

Case	Vol fraction of fuel in core F	k_∞	Percent Enrichment of uranium e	Vol of fuel in core cu ft V_f	Uranium in core kg M_u	Critical mass of U-235 M_{235}	Initial conversion Ratio R_c
1	0.05	1.05	1.30	395	22,100	298	.546
2		1.10	1.45	143	8,000	116	.492
3	0.075	1.05	1.25	427	23,900	298	.635
4		1.10	1.39	167	9,350	130	.600
5	0.10	1.05	1.275	445	24,900	318	.707
6		1.10	1.46	179	10,000	146	.668
7	0.15	1.05	1.525	474	26,600	405	.796
8		1.10	1.80	197	11,000	198	.780
9	0.20	1.05	2.24	520	29,100	652	.865
10		1.10	2.88	206	11,540	332	.865
11	0.25	1.05	4.36	575	32,200	1400	.900
12		1.10	7.05	240	13,430	952	.865

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