CHAPTER 14

NUCLEAR ASPECTS OF MOLTEN-SALT REACTORS*

The ability of certain molten salts to dissolve uranium and thorium salts in quantities of reactor interest made possible the consideration of fluid-fueled reactors with thorium in the fuel, without the danger of nuclear accidents as a result of the settling of a slurry. This additional degree of freedom has been exploited in the study of molten-salt reactors.

Mixtures of the fluorides of alkali metals and zirconium or beryllium, as discussed in Chapter 12, possess the most desirable combination of low neutron absorption, high solubility of uranium and thorium compounds, chemical inertness at high temperatures, and thermal and radiation stability. The following comparison of the capture cross sections of the alkali metals reveals that Li^7 containing 0.01% Li^6 has a cross section at 0.0795 ev and 1150°F that is a factor of 4 lower than that of sodium, which also has a relatively low cross section:

Element	*Cross section, barns*
Li^7 (containing 0.01% Li^6)	0.073
Sodium	0.290
Potassium	1.13
Rubidium	0.401
Cesium	29

The capture cross section of beryllium is also satisfactorily low at all neutron energies, and therefore mixtures of LiF and BeF_2, which have satisfactory melting points, viscosities, and solubilities for UF_4 and ThF_4, were selected for investigation in the reactor physics study.

Mixtures of NaF, ZrF_4, and UF_4 were studied previously, and such a fuel was successfully used in the Aircraft Reactor Experiment (see Chapters 12 and 16). Inconel was shown to be reasonably resistant to corrosion by this mixture at 1500°F, and there is reason to expect that Inconel equipment would have a life of at least several years at 1200°F. As a fuel for a central-station power reactor, however, the NaF–ZrF_4 system has several serious disadvantages. The sodium capture cross section is less favorable than that of Li^7. More important, recent data [1] indicate that the capture cross section of zirconium is quite high in the epithermal and intermediate neutron energy ranges. In comparison with the LiF–BeF_2 system, the NaF–ZrF_4 system has inferior heat-transfer characteristics.

*By L. G. Alexander.

Finally, the INOR alloys (see Chapter 13) show promise of being as resistant to the beryllium salts as to the zirconium salts, and therefore there is no compelling reason for selecting the $NaF–ZrF_4$ system.

Reactor calculations were performed by means of the Univac* program Ocusol [2], a modification of the Eyewash program [3], and the Oracle† program Sorghum. Ocusol is a 31-group, multiregion, spherically symmetric, age-diffusion code. The group-averaged cross sections for the various elements of interest that were used were based on the latest available data [4]. Where data were lacking, reasonable interpolations based on resonance theory were made. The estimated cross sections were made to agree with measured resonance integrals where available. Saturation and Doppler broadening of the resonances in thorium as a function of concentration were estimated. Inelastic scattering in thorium and fluorine was taken into account crudely by adjusting the value of $\xi\sigma_t$; however, the Ocusol code does not provide for group skipping or anisotropy of scattering.

Sorghum is a 31-group, two-region, zero-dimensional, burnout code. The group-diffusion equations were integrated over the core to remove the spatial dependency. The spectrum was computed, in terms of a space-averaged group flux, from group scattering and leakage parameters taken from an Ocusol calculation. A critical calculation requires about 1 min on the Oracle; changes in concentration of 14 elements during a specified time can then be computed in about 1 sec. The major assumption involved is that the group scattering and leakage probabilities do not change appreciably with changes in core composition as burnup progresses. This assumption has been verified to a satisfactory degree of approximation.

The molten salts may be used as homogeneous moderators or simply as fuel carriers in heterogeneous reactors. Although, as discussed below, graphite-moderated heterogeneous reactors have certain potential advantages, their technical feasibility depends upon the compatibility of fuel, graphite, and metal, which has not as yet been established. For this reason, the homogeneous reactors, although inferior in nuclear performance, have been given greatest attention.

A preliminary study indicated that if the integrity of the core vessel could be guaranteed, the nuclear economy of two-region reactors would probably be superior to that of bare and reflected one-region reactors. The two-region reactors were, accordingly, studied in detail. Although entrance and exit conditions dictate other than a spherical shape, it was necessary, for the calculations, to use a model comprising the following concentric

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*Universal Automatic Computer at New York University, Institute of Mathematics.

†Oak Ridge Automatic Computer and Logical Engine at Oak Ridge National Laboratory.

spherical regions: (1) the core, (2) an INOR–8 core vessel 1/3 in. thick, (3) a blanket approximately 2 ft thick, and (4) an INOR–8 reactor vessel 2/3 in. thick. The diameter of the core and the concentration of thorium in the core were selected as independent variables. The primary dependent variables were the critical concentration of the fuel (U^{235}, U^{233}, or Pu^{239}), and the distribution of the neutron absorptions among the various atomic species in the reactor. From these, the critical mass, critical inventory, regeneration ratio, burnup rate, etc. can be readily calculated, as described in the following section.

14–1. Homogeneous Reactors Fueled with U^{235}

While the isotope U^{233} would be a superior fuel in molten fluoride-salt reactors (see Section 14–2), it is unfortunately not available in quantity. Any realistic appraisal of the immediate capabilities of these reactors must be based on the use of U^{235}.

The study of homogeneous reactors was divided into two phases: (1) the mapping of the nuclear characteristics of the initial (i.e., "clean") states as a function of core diameter and thorium concentration, and (2) the analysis of the subsequent performance of selected initial states with various processing schemes and rates. The detailed results of these studies are given in the following paragraphs. Briefly, it was found that regeneration ratios of up to 0.65 can be obtained with moderate investment in U^{235} (less than 1000 kg) and that, if the fission products are removed (Article 14–1.2) at a rate such that the equilibrium inventory is equal to one year's production, the regeneration ratio can be maintained above 0.5 for at least 20 years.

14–1.1 Initial states. A complete parametric study of molten fluoride-salt reactors having diameters in the range of 4 to 10 ft and thorium concentrations in the fuel ranging from 0 to 1 mole % ThF_4 was performed. In these reactors, the basic fuel salt (fuel salt No. 1) was a mixture of 31 mole % BeF_2 and 69 mole % LiF, which has a density of about 2.0 g/cc at 1150°F. The core vessel was composed of INOR–8. The blanket fluid (blanket salt No. 1) was a mixture of 25 mole % ThF_4 and 75 mole % LiF, which has a density of about 4.3 g/cc at 1150°F. In order to shorten the calculations in this series, the reactor vessel was neglected, since the resultant error was small. These reactors contained no fission products or nonfissionable isotopes of uranium other than U^{238}.

A summary of the results is presented in Table 14–1, in which the neutron balance is presented in terms of neutrons absorbed in a given element per neutron absorbed in U^{235} (both by fission and the n–γ reaction). The sum of the absorptions is therefore equal to η, the number of neutrons produced by fission per neutron absorbed in fuel. Further, the sum of the

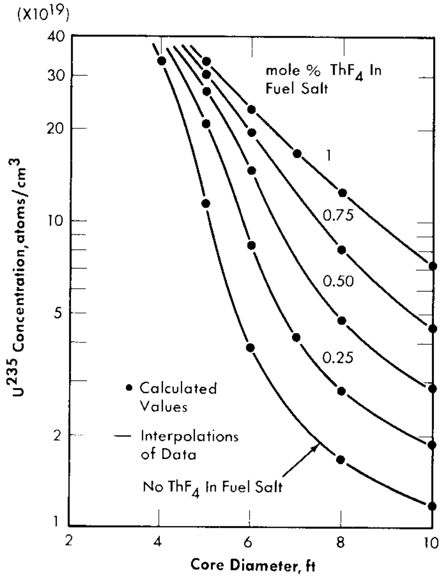

FIG. 14-1. Initial critical concentration of U^{235} in two-region, homogeneous, molten fluoride-salt reactors.

absorptions in U^{238} and thorium in the fuel, and in thorium in the blanket salt gives directly the regeneration ratio. The losses to other elements are penalties imposed on the regeneration ratio by these poisons; i.e., if the core vessel could be constructed of some material with a negligible cross section, the regeneration ratio could be increased by the amount listed for capture in the core vessel.

The inventories in these reactors depend in part on the volume of the fuel in the pipes, pumps, and heat exchangers in the external portion of the fuel circuit. The inventories listed in Table 14-1 are for systems having a volume of 339 ft^3 external to the core, which corresponds approximately to a power level of 600 Mw of heat. In these calculations it was assumed that the heat was transferred to an intermediate coolant composed of the fluorides of Li, Be, and Na before being transferred to sodium metal. In more recent designs (see Chapter 17), this intermediate salt loop has been replaced by a sodium loop, and the external volumes are somewhat less because of the improved equipment design and layout.

Critical concentration, mass, inventory, and regeneration ratio. The data in Table 14-1 are more easily comprehended in the form of graphs, such as Fig. 14-1, which presents the critical concentration in these reactors as a function of core diameter and thorium concentration in the fuel salt. The data points represent calculated values, and the lines are reasonable interpolations. The maximum concentration calculated, about 35×10^{19}

Table 14-1

Initial-State Nuclear Characteristics of Two-Region, Homogeneous, Molten Fluoride-Salt Reactors Fueled with U^{235}

Fuel salt No. 1: 31 mole % BeF_2 + 69 mole % LiF + UF_4 + ThF_4.
Blanket salt No. 1: 25 mole % ThF_4 + 75 mole % LiF.
Total power: 600 Mw (heat). External fuel volume: 339 ft^3.

Case number	1	2	3	4	5	6	7
Core diameter, ft	4	5	5	5	5	5	6
ThF_4 in fuel salt, mole %	0	0	0.25	0.5	0.75	1	0
U^{235} in fuel salt, mole %	0.952	0.318	0.561	0.721	0.845	0.938	0.107
U^{235} atom density*	33.8	11.3	20.1	25.6	30.0	33.3	3.80
Critical mass, kg of U^{235}	124	81.0	144	183	215	239	47.0
Critical inventory, kg of U^{235}	1380	501	891	1130	1330	1480	188
Neutron absorption ratios†							
U^{235} (fissions)	0.7023	0.7185	0.7004	0.6996	0.7015	0.7041	0.7771
U^{235} (n–γ)	0.2977	0.2815	0.2996	0.3004	0.2985	0.2959	0.2229
Be–Li–F in fuel salt	0.0551	0.0871	0.0657	0.0604	0.0581	0.0568	0.1981
Core vessel	0.0560	0.0848	0.0577	0.0485	0.0436	0.0402	0.1353
Li–F in blanket salt	0.0128	0.0138	0.0108	0.0098	0.0093	0.0090	0.0164
Leakage	0.0229	0.0156	0.0147	0.0143	0.0141	0.0140	0.0137
U^{238} in fuel salt	0.0430	0.0426	0.0463	0.0451	0.0431	0.0412	0.0245
Th in fuel salt			0.0832	0.1289	0.1614	0.1873	
Th in blanket salt	0.5448	0.5309	0.4516	0.4211	0.4031	0.3905	0.5312
Neutron yield, η	1.73	1.77	1.73	1.73	1.73	1.74	1.92
Median fission energy, ev	270	15.7	105	158	270	425	0.18
Thermal fissions, %	0.052	6.2	0.87	0.22	0.87	0.040	35
n–γ capture-to-fission ratio, α	0.42	0.39	0.43	0.43	0.43	0.4203	0.28
Regeneration ratio	0.59	0.57	0.58	0.60	0.61	0.62	0.56

continued

TABLE 14-1 (continued)

Case number	8	9	10	11	12	13	14
Core diameter, ft	6	6	6	6	7	8	8
ThF_4 in fuel salt, mole %	0.25	0.5	0.75	1	0.25	0	0.25
U^{235} in fuel salt, mole %	0.229	0.408	0.552	0.662	0.114	0.047	0.078
U^{235} atom density*	8.13	14.5	19.6	23.5	4.05	1.66	2.77
Critical mass, kg of U^{235}	101	179	243	291	79.6	48.7	81.3
Critical inventory, kg of U^{235}	404	716	972	1160	230	110	184
Neutron absorption ratios†							
U^{235} (fissions)	0.7343	0.7082	0.7000	0.7004	0.7748	0.8007	0.7930
U^{235} (n–γ)	0.2657	0.2918	0.3000	0.2996	0.2252	0.1993	0.2070
Be–Li–F in fuel salt	0.1082	0.0770	0.0669	0.0631	0.1880	0.4130	0.2616
Core vessel	0.0795	0.0542	0.0435	0.0388	0.0951	0.1491	0.1032
Li–F in blanket salt	0.0116	0.0091	0.0081	0.0074	0.0123	0.0143	0.0112
Leakage	0.0129	0.0122	0.0119	0.0116	0.0068	0.0084	0.0082
U^{238} in fuel salt	0.0375	0.0477	0.0467	0.0452	0.0254	0.0143	0.0196
Th in fuel salt	0.1321	0.1841	0.2142	0.2438	0.1761		0.2045
Th in blanket salt	0.4318	0.3683	0.3378	0.3202	0.4098	0.4073	0.3503
Neutron yield, η	1.82	1.75	1.73	1.73	1.91	2.00	1.96
Median fission energy, ev	5.6	38	100	120	0.16	Thermal	0.10
Thermal fissions, %	13	3	0.56	0.48	33	59	45
n-γ capture-to-fission ratio, α	0.36	0.41	0.42	0.42	0.29	0.25	0.26
Regeneration ratio	0.61	0.60	0.60	0.61	0.61	0.42	0.57

*Atoms ($\times 10^{-19}$)/cc. †Neutrons absorbed per neutron absorbed in U^{235}.

continued

TABLE 14–1 (continued)

Case number	15	16	17	18	19	20	21	22
Core diameter, ft	8	8	8	10	10	10	10	10
ThF_4 in fuel salt, mole %	0.5	0.75	1	0	0.25	0.5	0.75	1
U^{235} in fuel salt, mole %	0.132	0.226	0.349	0.033	0.052	0.081	0.127	0.205
U^{235} atom density*	4.67	8.03	12.4	1.175	1.86	2.88	4.50	7.28
Critical mass, kg of U^{235}	137	236	364	67.3	107	165	258	417
Critical inventory, kg of U^{235}	310	535	824	111	176	272	425	687
Neutron absorption ratios†								
U^{235} (fissions)	0.7671	0.7362	0.7146	0.8229	0.7428	0.7902	0.7693	0.7428
U^{235} (n–γ)	0.2329	0.2638	0.2854	0.1771	0.2572	0.2098	0.2307	0.2572
Be–Li–F in fuel salt	0.1682	0.1107	0.0846	0.5713	0.3726	0.2486	0.1735	0.1206
Core vessel	0.0722	0.0500	0.0373	0.1291	0.0915	0.0669	0.0497	0.0363
Li–F in blanket salt	0.0089	0.0071	0.0057	0.0114	0.0089	0.0073	0.0060	0.0049
Leakage	0.0080	0.0077	0.0074	0.0061	0.0060	0.0059	0.0057	0.0055
U^{238} in fuel salt	0.0272	0.0368	0.0428	0.0120	0.0153	0.0209	0.0266	0.0343
Th in fuel salt	0.3048	0.3397	0.3515		0.2409	0.3691	0.4324	0.4506
Th in blanket salt	0.3056	0.2664	0.2356	0.3031	0.2617	0.2332	0.2063	0.1825
Neutron yield, η	1.89	1.82	1.76	2.03	2.00	1.95	1.90	1.83
Median fission energy, ev	0.17	5.3	27	Thermal	Thermal	0.100	0.156	1.36
Thermal fissions, %	29	13	5	66	56	43	30	16
n–γ capture-to-fission ratio, α	0.30	0.36	0.40	0.21	0.24	0.26	0.30	0.35
Regeneration ratio	0.64	0.64	0.63	0.32	0.52	0.62	0.67	0.67

*Atoms ($\times 10^{-19}$)/cc.

†Neutrons absorbed per neutron absorbed in U^{235}.

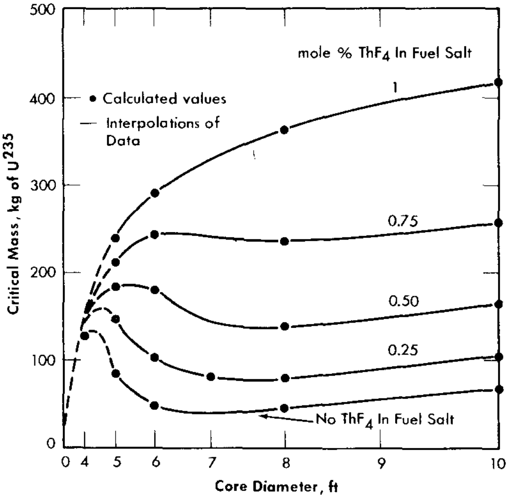

FIG. 14-2. Initial critical masses of U^{235} in two-region, homogeneous, molten fluoride-salt reactors.

atoms of U^{235} per cubic centimeter of fuel salt, or about 1 mole % UF_4, is an order of magnitude smaller than the maximum permissible concentration (about 10 mole %).

The corresponding critical masses are graphed in Fig. 14-2. As may be seen, the critical mass is a rather complex function of the diameter and the thorium concentration. The calculated points are shown here also, and the solid lines represent, it is felt, reliable interpolations. The dashed lines were drawn where insufficient numbers of points were calculated to define the curves precisely; however, they are thought to be qualitatively correct. Since reactors having diameters less than 6 ft are not economically attractive, only one case with a 4-ft-diameter core was computed.

The critical masses obtained in this study ranged from 40 to 400 kg of U^{235}. However, the critical inventory in the entire fuel circuit is of more interest to the reactor designer than is the critical mass. The critical inventories corresponding to an external fuel volume of 339 ft^3 are therefore shown in Fig. 14-3. Inventories for other external volumes may be computed from the relation

$$I = M\left(1 + \frac{6V_e}{D^3}\right),$$

where D is the core diameter in feet, M is the critical mass taken from Fig. 14-2, V_e is the volume of the external system in cubic feet, and I is the inventory in kilograms of U^{235}. The inventories plotted in Fig. 14-3

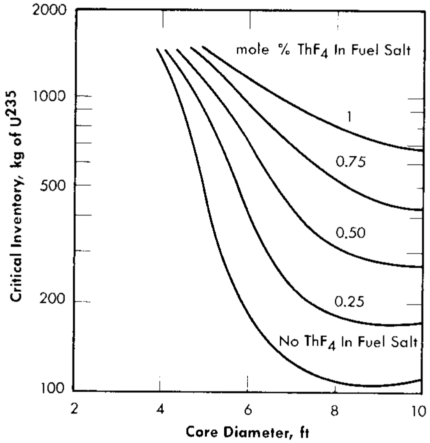

FIG. 14-3. Initial critical inventories of U^{235} in two-region, homogeneous, molten fluoride-salt reactors. External fuel volume, 339 ft^3.

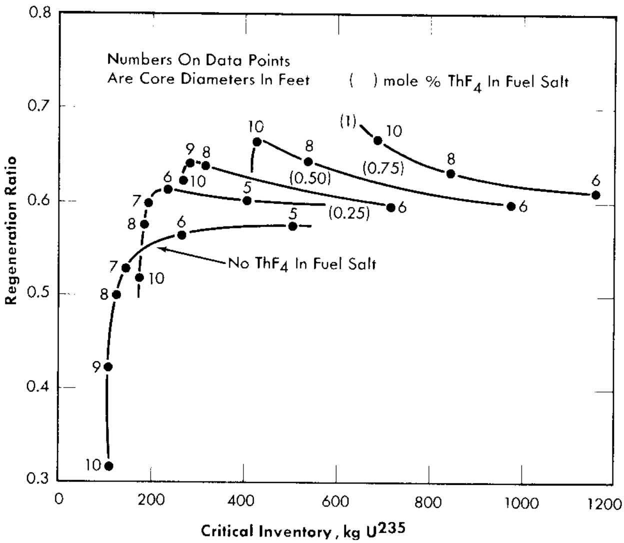

FIG. 14-4. Initial fuel regeneration in two-region, homogeneous, molten fluoride-salt reactors fueled with U^{235}. Total power, 600 Mw (heat); external fuel volume, 339 ft^3; core and blanket salts No. 1.

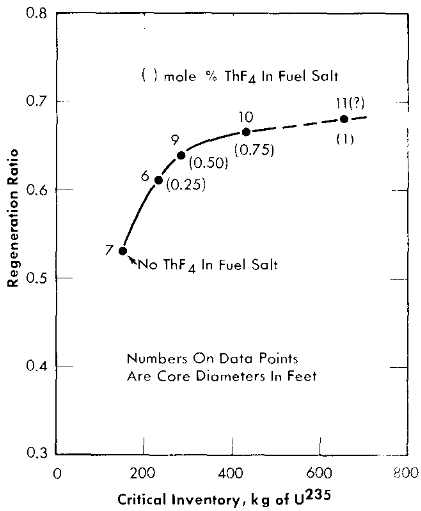

FIG. 14-5. Maximum initial regeneration ratios in two-region, homogeneous, molten fluoride-salt reactors fueled with U^{235}. Total power, 600 Mw (heat); external fuel volume, 339 ft^3.

range from slightly above 100 kg in an 8-ft-diameter core with no thorium present to 1500 kg in a 5-ft-diameter core with 1 mole % ThF_4 present.

The optimum combination of core diameter and thorium concentration is, qualitatively, that which minimizes the sum of inventory charges (including charges on Li^7, Be, and Th) and fuel reprocessing costs. The fuel costs are directly related to the regeneration ratio, and this varies in a complex manner with inventory of U^{235} and thorium concentration, as shown in Fig. 14-4. It may be seen that at a given thorium concentration, the regeneration ratio (with one exception) passes through a maximum as the core diameter is varied between 5 and 10 ft. These maxima increase with increasing thorium concentration, but the inventory values at which they occur also increase.

Plotting the maximum regeneration ratio versus critical inventory generates the curve shown in Fig. 14-5. It may be seen that a small investment in U^{235} (200 kg) will give a regeneration ratio of 0.58, that 400 kg will give a ratio of 0.66, and that further increases in fuel inventory have little effect.

The effects of changes in the compositions of the fuel and blanket salts are indicated in the following description of the results of a series of calculations for which salts with more favorable melting points and viscosities were assumed. The BeF_2 content was raised to 37 mole % in the fuel salt

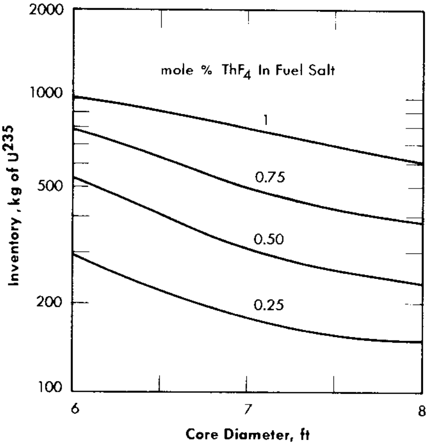

FIG. 14-6. Initial critical inventories of U^{235} in two-region, homogeneous, molten fluoride-salt reactors. Total power, 600 Mw (heat); external fuel volume, 339 ft^3; core and blanket salts No. 2.

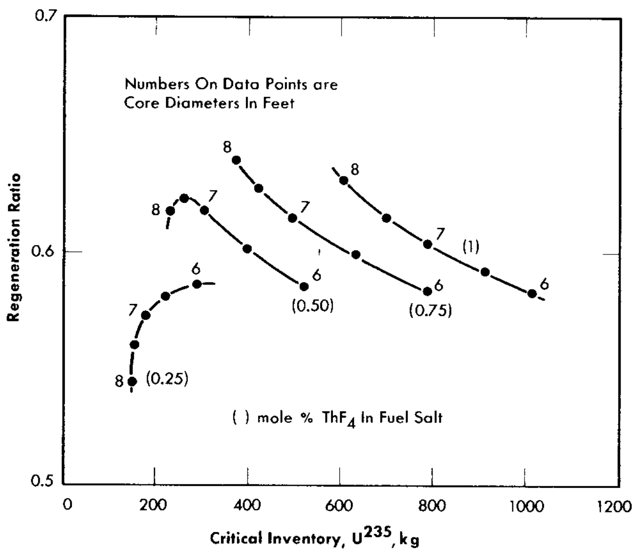

FIG. 14-7. Initial fuel regeneration in two-region, homogeneous, molten fluoride-salt reactors fueled with U^{235}. Total power, 600 Mw (heat); external fuel volume, 339 ft^3; core and blanket salts No. 2.

(fuel salt No. 2), and the blanket composition (blanket salt No. 2) was fixed at 13 mole % ThF_4, 16 mole % BeF_2, and 71 mole % LiF. Blanket salt No. 2 is a somewhat better reflector than No. 1, and fuel salt No. 2 a somewhat better moderator. As a result, at a given core diameter and thorium concentration in the fuel salt, both the critical concentration and the regeneration ratio are somewhat lower for the No. 2 salts.

Reservations concerning the feasibility of constructing and guaranteeing the integrity of core vessels in large sizes (10 ft and over), together with preliminary consideration of inventory charges for large systems, led to the conclusion that a feasible reactor would probably have a core diameter lying in the range between 6 and 8 ft. Accordingly, a parametric study in this range with the No. 2 fuel and blanket salts was performed. In this study the presence of an outer reactor vessel consisting of 2/3 in. of INOR-8 was taken into account. The results are presented in Table 14-2 and Figs. 14-6 and 14-7. In general, the nuclear performance is somewhat better with the No. 2 salt than with the No. 1 salt.

Neutron balances and miscellaneous details. The distributions of the neutron captures are given in Tables 14-1 and 14-2, where the relative hardness of the neutron spectrum is indicated by the median fission energies and the percentages of thermal fissions. It may be seen that losses to Li, Be, and F in the fuel salt and to the core vessel are substantial, especially in the more thermal reactors (e.g., Case No. 18). However, in the thermal reactors, losses by radiative capture in U^{235} are relatively low. Increasing the hardness decreases losses to salt and core vessel sharply (Case No. 5), but increases the loss to the n-γ reaction. It is these opposing trends which account for the complicated relation between regeneration ratio and critical inventory exhibited in Figs. 14-4 and 14-7. The numbers given for capture in the Li and F in the blanket show that these elements are well shielded by the thorium in the blanket, and the leakage values show that leakage from the reactor is less than 0.01 neutron per neutron absorbed in U^{235} in reactors over 6 ft in diameter. The blanket contributes substantially to the regeneration of fuel, accounting for not less than one-third of the total even in the 10-ft-diameter core containing 1 mole % ThF_4.

Effect of substitution of sodium for Li^7. In the event that Li^7 should prove not to be available in quantity, it would be possible to operate the reactor with mixtures of sodium and beryllium fluorides as the basic fuel salt. The penalty imposed by sodium in terms of critical inventory and regeneration ratio is shown in Fig. 14-8, where typical Na-Be systems are compared with the corresponding Li-Be systems. With no thorium in the core, the use of sodium increases the critical inventory by a factor of 1.5 (to about 300 kg) and lowers the regeneration ratio by a factor of 2. The regeneration penalty is less severe, percentagewise, with 1 mole % ThF_4 in the fuel salt; in an 8-ft-diameter core, the inventory rises from 800 kg to 1100 kg

TABLE 14–2

INITIAL-STATE NUCLEAR CHARACTERISTICS OF TWO-REGION, HOMOGENEOUS, MOLTEN FLUORIDE-SALT REACTORS FUELED WITH U^{235}

Fuel salt No. 2: 37 mole % BeF_2 + 63 mole % LiF + UF_4 + ThF_4.
Blanket salt No. 2: 13 mole % ThF_4 + 16 mole % BeF_2 + 71 mole % LiF.
Total power: 600 Mw (heat). External fuel volume: 339 ft^3.

Case number	23	24	25	26	27	28
Core diameter, ft	6	6	6	6	7	7
ThF_4 in fuel salt, mole %	0.25	0.5	0.75	1	0.25	0.5
U^{235} in fuel salt, mole %	0.169	0.310	0.423	0.580	0.084	0.155
U^{235} atom density*	5.87	10.91	15.95	20.49	3.13	5.38
Critical mass, kg of U^{235}	72.7	135	198	254	61.5	106
Critical inventory, kg of U^{235}	291	540	790	1010	178	306
Neutron absorption ratios†						
U^{235} (fissions)	0.7516	0.7174	0.7044	0.6958	0.7888	0.7572
U^{235} (n–γ)	0.2484	0.2826	0.2956	0.3042	0.2112	0.2428
Be–Li–F in fuel salt	0.1307	0.0900	0.0763	0.0692	0.2147	0.1397
Core vessel	0.1098	0.0726	0.0575	0.0473	0.1328	0.0905
Li–F in blanket salt	0.0214	0.0159	0.0132	0.0117	0.0215	0.0167
Outer vessel	0.0024	0.0021	0.0021	0.0019	0.0019	0.0018
Leakage	0.0070	0.0065	0.0064	0.0061	0.0052	0.0050
U^{238} in fuel salt	0.0325	0.0426	0.0452	0.0477	0.0214	0.0307
Th in fuel salt	0.1360	0.1902	0.2212	0.2387	0.1739	0.2565
Th in blanket salt	0.4165	0.3521	0.3178	0.2962	0.3770	0.3294
Neutron yield, η	1.86	1.77	1.74	1.72	1.95	1.87
Median fission energy, ev	0.480	10.47	58.10	76.1	0.1223	0.415
Thermal fissions, %	21	7	2.8	0.84	43	24
n–γ capture-to-fission ratio, α	0.33	0.39	0.42	0.44	0.37	0.32
Regeneration ratio	0.59	0.58	0.58	0.58	0.57	0.62

continued

TABLE 14-2 (continued)

Case number	29	30	31	32	33	34
Core diameter, ft	7	7	8	8	8	8
ThF_4 in fuel salt, mole %	0.75	1	0.25	0.5	0.75	1
U^{235} in fuel salt, mole %	0.254	0.366	0.064	0.099	0.163	0.254
U^{235} atom density*	8.70	13.79	2.24	3.51	5.62	9.09
Critical mass, kg of U^{235}	171	271	65.7	103	165	267
Critical inventory, kg of U^{235}	494	783	149	233	374	604
Neutron absorption ratios†						
U^{235} (fissions)	0.7282	0.7094	0.8014	0.7814	0.7536	0.7288
U^{235} (n–γ)	0.2718	0.2906	0.1986	0.2186	0.2464	0.2712
Be–Li–F in fuel salt	0.1010	0.0824	0.2769	0.1945	0.1354	0.1016
Core vessel	0.0644	0.0497	0.1308	0.0967	0.0696	0.0518
Li–F in blanket salt	0.0131	0.0108	0.0198	0.0162	0.0130	0.0105
Outer vessel	0.0016	0.0015	0.0017	0.0016	0.0014	0.0013
Leakage	0.0048	0.0045	0.0045	0.0043	0.0042	0.0040
U^{238} in fuel salt	0.0392	0.0447	0.0177	0.0233	0.0315	0.0392
Th in fuel salt	0.2880	0.3022	0.1978	0.3043	0.3501	0.3637
Th in blanket salt	0.2866	0.2566	0.3240	0.2892	0.2561	0.2280
Neutron yield, η	1.80	1.75	1.97	1.93	1.86	1.80
Median fission energy, ev	7.61	25.65	51% thermal	0.136	0.518	7.75
Thermal fissions, %	11	4.3	51	38	23	11
n–γ capture-to-fission ratio, α	0.37	0.41	0.25	0.28	0.33	0.37
Regeneration ratio	0.61	0.60	0.54	0.62	0.64	0.63

*Atoms ($\times 10^{-19}$)/cc.

†Neutrons absorbed per neutron absorbed in U^{235}.

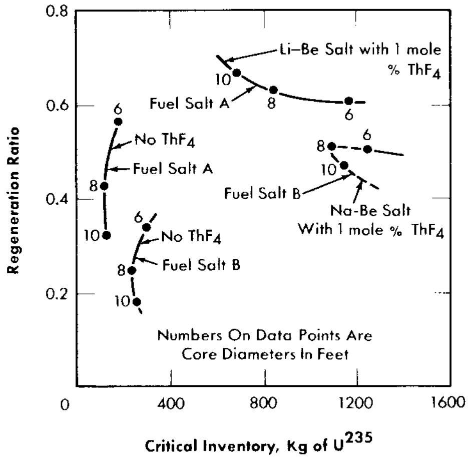

FIG. 14–8. Comparison of regeneration ratio and critical inventory in two-region, homogeneous, molten fluoride-salt reactors fueled with U^{235}. Fuel salt A: 37 mole % BeF_2 plus 63 mole % Li^7F. Fuel salt B: 46 mole % BeF_2 plus 54 mole % NaF.

and the regeneration ratio falls from 0.62 to 0.50. Details of the neutron balances are given in Table 14–3.

Reactivity coefficients. By means of a series of calculations in which the thermal base, the core radius, and the density of the fuel salt are varied independently, the components of the temperature coefficient of reactivity of a reactor can be estimated as illustrated below for a core 8 ft in diameter and a thorium concentration of 0.75 mole % in the fuel salt at 1150°F. From the expression

$$k = f(T, \rho, R),$$

where k is the multiplication constant, T is the mean temperature in the core, ρ is the mean density of the fuel salt in the core, and R is the core radius, it follows that

$$\frac{1}{k}\frac{dk}{dT} = \frac{1}{k}\left(\frac{\partial k}{\partial T}\right)_{\rho,\,R} + \frac{1}{k}\left(\frac{\partial k}{\partial R}\right)_{\rho,\,T}\frac{dR}{dT} + \frac{1}{k}\left(\frac{\partial k}{\partial \rho}\right)_{R,\,T}\frac{d\rho}{dT}, \tag{14–1}$$

where the term $(1/k)(\partial k/\partial T)_{\rho,\,R}$ represents the fractional change in k due to a change in the thermal base for slowing down of neutrons, the term $(1/k)(\partial k/\partial \rho)_{R,\,T}$ represents the change due to expulsion of fuel from the core by thermal expansion of the fluid, and the term $(1/k)(\partial k/\partial R)_{\rho,\,T}$ represents the change due to an increase in core volume and fuel holding

Table 14-3

Initial Nuclear Characteristics of Two-Region, Homogeneous, Molten Sodium-Beryllium Fluoride Reactors Fueled with U^{235}

Fuel salt: 53 mole % NaF + 46 mole % BeF_2 + 1 mole % (ThF_4 + UF_4).
Blanket salt: 58 mole % NaF + 35 mole % BeF_2 + 7 mole % ThF_4.
Total power: 600 Mw (heat). External fuel volume: 339 ft^3.

Case number	35	36	37	38	39	40
Core diameter, ft	6	6	8	8	10	10
ThF_4 in fuel salt, mole %	0	1	0	1	0	1
U^{235} in fuel salt, mole %	0.174	0.7014	0.091	0.465	0.070	0.282
U^{235} atom density*	6.17	24.9	3.24	16.5	2.47	124.0
Critical mass, kg of U^{235}	76.4	308	95.1	484	142	710
Critical inventory, kg of U^{235}	306	1230	215	1100	234	1170
Neutron absorption ratios†						
U^{235} (fissions)	0.7417	0.6986	0.7737	0.7011	0.7862	0.7081
U^{235} (n-γ)	0.2583	0.3014	0.2263	0.2989	0.2138	0.2919
Na–Be–F in fuel salt	0.2731	0.1153	0.4755	0.1411	0.6119	0.2306
Core vessel	0.1181	0.0476	0.1125	0.0392	0.0917	0.2306
Na–Be–F in blanket salt	0.0821	0.0431	0.0660	0.0315	0.0495	0.2306
Leakage	0.0222	0.0182	0.0145	0.0116	0.0105	0.2306
U^{238} in fuel salt	0.0360	0.0477	0.0263	0.0484	0.0232	0.0467
Th in fuel salt		0.2418		0.3150		0.3670
Th in blanket salt	0.3004	0.2120	0.2163	0.1450	0.1550	0.1048
Neutron yield, η	1.83	1.73	1.91	1.73	1.94	1.75
Median fission energy, ev	1.3	190	0.20	36	0.087	
Thermal fissions, %	17	0.42	34	1.4	4.1	
n–γ capture-to-fission ratio, α	0.25	0.43	0.29	0.43	0.27	0.41
Regeneration ratio	0.34	0.50	0.24	0.51	0.18	0.52

*Atoms ($\times 10^{-19}$)/cc.

†Neutrons absorbed per neutron absorbed in U^{235}.

capacity. The coefficient dR/dT may be related to the coefficient for linear expansion, α, of INOR-8, viz:

$$\frac{dR}{dT} = R\alpha.$$

Likewise the term $d\rho/dT$ may be related to the coefficient of cubical expansion, β, of the fuel salt:

$$\frac{d\rho}{dT} = -\rho\beta.$$

From the nuclear calculations, the components of the temperature coefficient were estimated, as follows:

$$\frac{1}{k}\left(\frac{\partial k}{\partial T}\right)_{\rho,\,R} = -(0.13 \pm 0.02) \times 10^{-5}/{}^\circ\mathrm{F},$$

$$\frac{R}{k}\left(\frac{\partial k}{\partial R}\right)_{\rho,\,T} = +\,0.412 \pm 0.0005,$$

$$\frac{\rho}{k}\left(\frac{\partial k}{\partial \rho}\right)_{R,\,T} = -0.405 \pm 0.0005.$$

The linear coefficient of expansion, α, of INOR-8 was estimated to be $(8.0 \pm 0.5) \times 10^{-6}/{}^\circ\mathrm{F}$ [5], and the coefficient of cubical expansion, β, of the fuel was estimated to be $(9.889 \pm 0.005) \times 10^{-5}/{}^\circ\mathrm{F}$ from a correlation of the density given by Powers [6]. Substitution of these values in Eq. (14–1) gives

$$\frac{1}{k}\frac{dk}{dT} = -(3.80 \pm 0.04) \times 10^{-5}/{}^\circ\mathrm{F}$$

for the temperature coefficient of reactivity of the fuel. In this calculation, the effects of changes with temperature in Doppler broadening and saturation of the resonances in Th and U^{235} were not taken into account. Since the effective widths of the resonances would be increased at higher temperatures, the thorium would contribute a reactivity decrease and the U^{235} an increase. These effects are thought to be small, and they tend to cancel each other.

Additional coefficients of interest are those for U^{235} and thorium. For the 8-ft-diameter cores,

$$\frac{N(U^{235})}{k}\left(\frac{\partial k}{\partial N(U^{235})}\right)_{N(\mathrm{Th})} = \frac{1 + [0.17N_c(U^{235}) \times 10^{-19}]}{2.47N_c(U^{235}) \times 10^{-19}}$$

and

$$\frac{N(\mathrm{Th})}{k}\left(\frac{\partial k}{\partial N(\mathrm{Th})}\right)_{N(U^{235})} = \frac{N(\mathrm{Th})}{k}\left(\frac{\partial k}{\partial N(U^{235})}\right)_{N(\mathrm{Th})}\frac{dN_c(U^{235})}{dN(\mathrm{Th})},$$

where

$$\frac{dN_c(U^{235})}{dN(\mathrm{Th})} = 0.0805\, e^{0.0595N(\mathrm{Th}) \times 10^{-19}}.$$

In these equations, $N(U^{235})$ represents the atomic density of U^{235} in atoms per cubic centimeter, $N_c(U^{235})$ is the critical density of U^{235}, and $N(\mathrm{Th})$ is the density of thorium atoms.

Heat release in core vessel and blanket. The core vessel of a molten-salt reactor is heated by gamma radiation emanating from the core and blanket and from within the core vessel itself. Estimates of the gamma heating can be obtained by detailed analyses of the type illustrated by Alexander and Mann [7]. The gamma-ray heating in the core vessel of a reactor with an 8-ft-diameter core and 0.5 mole % ThF_4 in the fuel salt has been estimated to be the following:

Source	*Heat release rate, w/cm³*
Radioactive decay in core	1.4
Fission, n–γ capture, and inelastic scattering in core	5.2
n–γ capture in core vessel	4.5
n–γ capture in blanket	0.3
Total	11.4

Estimates of gamma-ray source strengths can be used to provide a crude estimate of the gamma-ray current entering the blanket. For the 8-ft-diameter core, the core contributes 45.3 w of gamma energy per square centimeter to the blanket, and the core vessel contributes 6.8 w/cm², which, multiplied by the surface area of the core vessel, gives a total energy escape into the blanket of 9.7 Mw. Some of this energy will be reflected into the core, of course, and some will escape from the reactor vessel, and

therefore the value of 9.7 Mw is an upper limit. To this may be added the heat released by capture of neutrons in the blanket. From the Ocusol-A calculation for the 8-ft-diameter core and a fuel salt containing 0.5 mole % ThF_4 it was found that 0.176 of the neutrons would be captured in the blanket. If an energy release of 7 Mev/capture is assumed, the heat release at a power level of 600 Mw (heat) is estimated to be 8.6 Mw. The total is thus 18.3 Mw or, say, 20 ± 5 Mw, to allow for errors.

No allowance was made for fissions in the blanket. These would add 6 Mw for each 1% of the fissions occurring in the blanket. Thus it appears that the heat release rate in the blanket might range up to 50 Mw.

14–1.2 Intermediate states. *Without reprocessing of fuel salt.* The nuclear performance of a homogeneous molten-salt reactor changes during operation at power because of the accumulation of fission products and nonfissionable isotopes of uranium. It is necessary to add U^{235} to the fuel salt to overcome these poisons and, as a result, the neutron spectrum is hardened and the regeneration ratio decreases because of the accompanying decrease in η for U^{235} and the increased competition for neutrons by the poisons relative to thorium. The accumulation of the superior fuel U^{233} compensates for these effects only in part. The decline in the regeneration ratio and the increase in the critical inventory during the first year of operation of three reactors having 8-ft-diameter cores charged, respectively, with 0.25, 0.75, and 1 mole % ThF_4 are illustrated in Fig. 14–9. The critical inventory increases by about 300 kg, and the regeneration ratio falls about 16%. The gross burnup of fuel in the reactor charged with 1 mole % ThF_4 and operated at 600 Mw with a load factor of 0.80% amounts to about 0.73 kg/day. The U^{235} burnup falls from this value as U^{233} assumes part of the load. During the first month of operation, the U^{235} burnup averages 0.69 kg/day. Overcoming the poisons requires 1.53 kg more and brings the feed rate to 2.22 kg/day. The initial rate is high because of the holdup of bred fuel in the form of Pa^{233}. As the concentration of this isotope approaches equilibrium, the U^{235} feed rate falls rapidly. At the end of the first year the burnup rate has fallen to 0.62 kg/day and the feed rate to 1.28 kg/day. At this time U^{233} contributes about 12% of the fissions. The reactor contains 893 kg of U^{235}, 70 kg of U^{233}, 7 kg of Pu^{239}, 62 kg of U^{236}, and 181 kg of fission products. The U^{236} and the fission products capture 1.8 and 3.8% of all neutrons and impair the regeneration ratio by 0.10 units. Details of the inventories and concentrations are given in Table 14–4.

With reprocessing of fuel salt. If the fission products were allowed to accumulate indefinitely, the fuel inventory would become prohibitively large and the neutron economy would become very poor. However, if the fission products are removed, as described in Chapter 12, at a rate such that the

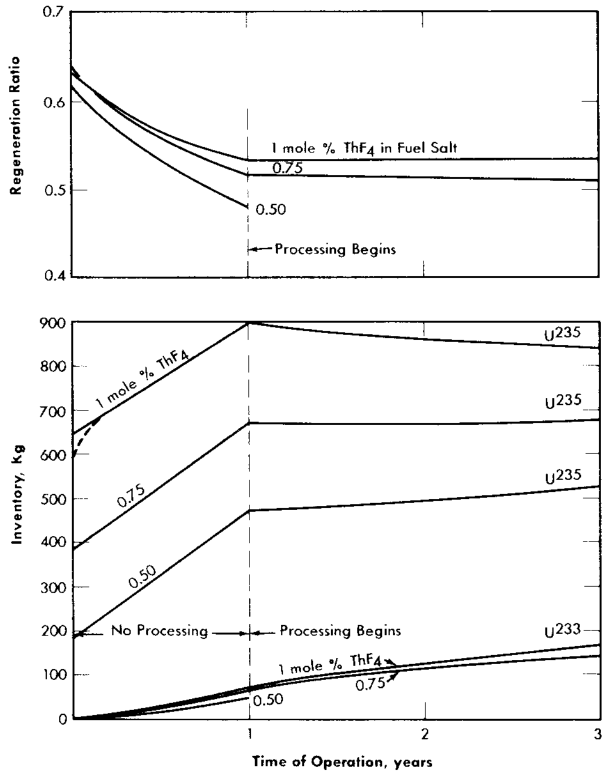

FIG. 14-9. Operating performance of two-region, homogeneous, molten fluoride-salt reactors fueled with U^{235}. Core diameter, 8 ft; total power, 600 Mw (heat); load factor, 0.80.

equilibrium inventory is, for example, equal to the first year's production, then the increase in U^{235} inventory and the decrease in regeneration ratio are effectively arrested, as shown in Fig. 14-10. The fuel-addition rate drops immediately from 1.28 to 0.73 kg/day when processing is started. At the end of two years, the addition rate is down to 0.50 kg/day, and it continues to decline slowly to 0.39 kg/day after 20 years of operation. The nonfissionable isotopes of uranium continue to accumulate, of course, but these are nearly compensated by the ingrowth of U^{233}. As shown in Fig. 14-10, the inventory of U^{235} actually decreases for several years in a typical case, and then increases only moderately during a lifetime of 20 years.

The rapid increase in critical inventory of U^{235} during the first year can be avoided by partial withdrawl of thorium. In Fig. 14-10 the dashed lines indicate the course of events when thorium is removed at the rate of 1/900 per day. Burnup reduces the thorium concentration by another

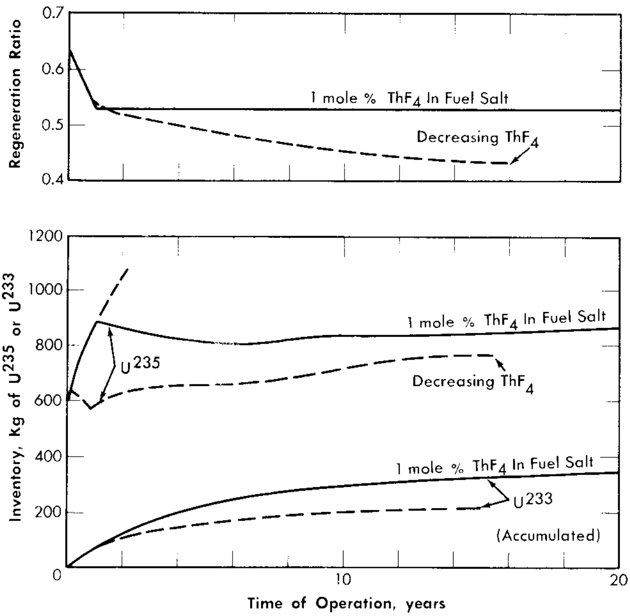

FIG. 14–10. Long-term nuclear performance of typical two-region, homogeneous, molten fluoride-salt reactors fueled with U^{235}. Core diameter, 8 ft; total power, 600 Mw (heat); load factor, 0.80.

1/4300 per day. The U^{235} inventory rises to 826 kg and then falls, at the end of eight months, to 587 kg. At this time, the processing rate is increased to 1/240 per day (eight-month cycle), but the thorium is returned to the core and the thorium concentration falls thereafter only by burnup. It may be seen that the U^{235} inventory creeps up slowly and that the regeneration ratio falls slowly. The increase in U^{235} inventory could have been prevented by withdrawing thorium at a small rate; however, the regeneration ratio would have fallen somewhat more rapidly, and more U^{235} feed would have been required to compensate for burnup.

14–2. HOMOGENEOUS REACTORS FUELED WITH U^{233}

Uranium–233 is a superior fuel for use in molten fluoride-salt reactors in almost every respect. The fission cross section in the intermediate range of neutron energies is greater than the fission cross sections of U^{235} and Pu^{239}. Thus initial critical inventories are less, and less additional fuel is required to override poisons. Also, the parasitic cross section is substantially less, and fewer neutrons are lost to radiative capture. Further, the radiative captures result in the immediate formation of a fertile iso-

TABLE 14-4

NUCLEAR PERFORMANCE OF A TWO-REGION, HOMOGENEOUS, MOLTEN FLUORIDE-SALT REACTOR FUELED WITH U^{235} AND CONTAINING 1 MOLE % ThF_4 IN THE FUEL SALT

Core diameter: 8 ft. Total power: 600 Mw (heat).
External fuel volume: 339 ft³. Load factor: 0.8.

	Initial state			After 1 year		
	Inventory, kg	Absorptions, %	Fissions, %	Inventory, kg	Absorptions, %	Fissions, %
Core elements						
Th^{232}	2,100	20.3		2,100	16.7	
Pa^{233}				8.2	0.3	
U^{233}				61.0	5.9	12.5
U^{234}				1.9	0.0	
U^{235}	604	55.4	100	893	49.3	86.3
U^{236}				62.2	1.8	
Np^{237}				4.2	0.2	
U^{238}	45.3	2.2		57.9	2.0	
Pu^{239}				6.8	0.8	1.2
Fission fragments				181	3.8	
Li^7	3,920	1.9		3,920	0.9	
Be^9	3,008	0.6		3,008	0.5	
F^{19}	24,000	3.2		24,000	3.0	
Blanket element						
U^{233}				8.7		
Total fuel	604			963		
U^{235} burnup rate, kg/day	0.69			0.62		
U^{235} feed rate, kg/day	2.22			1.28–0.73		
Regeneration ratio	0.64			0.53		

continued

Table 14-4 (continued)

	After 2 years			After 5 years		
	Inventory, kg	Absorptions, %	Fissions, %	Inventory, kg	Absorptions, %	Fissions, %
Core elements						
Th^{232}	2,100	16.3		2,100	15.4	
Pa^{233}	7.9	0.2		7.5	0.2	
U^{233}	110	9.7	20.8	201	15.3	33.0
U^{234}	6.5	0.1		27.1	0.4	
U^{235}	863	44.3	77.4	818	36.9	64.1
U^{236}	115	3.1		222	5.2	
Np^{237}	0.8	0.4		1.8	0.8	
U^{238}	69.7	2.3		9.0	2.7	
Pu^{239}	12.0	1.3	1.8	24.3	2.0	2.9
Fission fragments	181	3.6		181	3.1	
Li^7	3,920	0.8		3,920	0.6	
Be^9	3,008	0.5		3,008	0.5	
F^{19}	24,000	3.0		24,000	3.0	
Blanket element						
U^{233}	16			24		
Total fuel	990			1,045		
U^{235} burnup rate, kg/day	0.58			0.47		
U^{235} feed rate, kg/day	0.50			0.45		
Regeneration ratio	0.53			0.54		

continued

TABLE 14-4 (continued)

	After 10 years			After 20 years		
	Inventory, kg	Absorptions, %	Fissions, %	Inventory, kg	Absorptions, %	Fissions, %
Core elements						
Th^{232}	2,100	14.6		2,100	13.7	
Pa^{233}	7.1	0.2		6.7	0.2	
U^{233}	266	17.6	38.3	322	18.8	41.0
U^{234}	64.4	0.8		124	1.4	
U^{235}	831	33.5	58.2	872	31.7	54.9
U^{236}	328	6.7		450	7.9	
Np^{237}	2.6	0.9		3.2	1.0	
U^{238}	10.8	2.9		12.9	3.0	
Pu^{239}	37.3	2.4	3.5	52.6	2.8	4.1
Fission fragments	181	2.7		181	2.4	
Li^7	3,920	0.5		3,920	0.4	
Be^9	3,008	0.5		3,008	0.5	
F^{19}	24,000	3.0		24,000	3.0	
Blanket element						
U^{233}	28			33		
Total fuel	1,129			1,232		
U^{235} burnup rate, kg/day	0.41			0.38		
U^{235} feed rate, kg/day	0.44			0.39		
Regeneration ratio	0.533			0.530		

tope, U^{234}. The rate of accumulation of U^{236} is orders of magnitude smaller than with U^{235} as a fuel, and buildup of Np^{237} and Pu^{239} is negligible.

The mean neutron energy is rather nearer to thermal in these reactors than it is in the corresponding U^{235} cases. Consequently, losses to core vessel and to core salt tend to be higher. Both losses will be reduced substantially at higher thorium concentrations.

14–2.1 Initial states. Results from a parametric study of the nuclear characteristics of two-region, homogeneous, molten fluoride-salt reactors fueled with U^{233} are given in Table 14–5. The core diameters considered range from 3 to 10 ft, and the thorium concentrations range from 0.25 to 1 mole %. Although the regeneration ratios are less than unity, they are very good compared with those obtained with U^{235}. With 1 mole % ThF_4 in an 8-ft-diameter core, the U^{233} inventory was only 196 kg, and the regeneration ratio was 0.91.

The regeneration ratios and fuel inventories of reactors of various diameters containing 0.25 mole % thorium and fueled with U^{235} or U^{233} are compared in Fig. 14–11. The superiority of U^{233} is obvious.

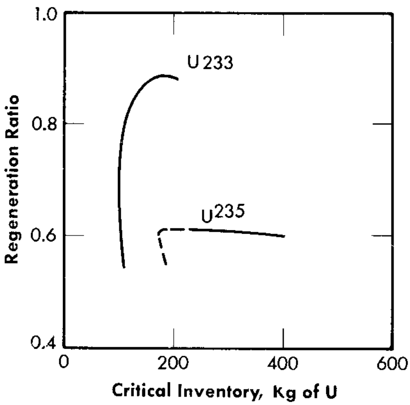

FIG. 14–11. Comparison of regeneration ratios in molten-salt reactors containing 0.25 mole % ThF_4 and U^{235}- or U^{233}-enriched fuel.

14–2.2 Intermediate states. Calculations of the long-term performance of one reactor (Case 51, Table 14–5) with U^{233} as the fuel are described below. The core diameter used was 8 ft and the thorium concentration was 0.75 mole %. The changes in inventory of U^{233} and regeneration ratio are listed in Table 14–6. During the first year of operation, the inventory rises from 129 to 199 kg, and the regeneration ratio falls from 0.82 to 0.71. If the reprocessing required to hold the concentration of fission products

TABLE 14–5

NUCLEAR CHARACTERISTICS OF TWO-REGION, HOMOGENEOUS, MOLTEN FLUORIDE-SALT REACTORS FUELED WITH U^{233}

Core diameter: 8 ft. Total power: 600 Mw (heat).
External fuel volume: 339 ft³. Load factor: 0.8.

Case number	41	42	43	44	45	46
Fuel and blanket salts*	1	1	1	1	1	1
Core diameter, ft	3	4	4	5	6	6
ThF_4 in fuel salt, mole %	0	0	0.25	0	0.25	0.25
U^{233} in fuel salt, mole %	0.592	0.158	0.233	0.106	0.048	0.066
U^{233} atom density†	21.0	6.09	8.26	3.75	1.66	2.36
Critical mass, kg of U^{233}	64.9	22.3	30.3	26.9	20.5	29.2
Critical inventory, kg of U^{233}	1620	248	337	166	82.0	117
Neutron absorption ratios‡						
U^{233} (fissions)	0.8754	0.8706	0.8665	0.8725	0.8814	0.8779
U^{233} (n–γ)	0.1246	0.1294	0.1335	0.1275	0.1186	0.1221
Be–Li–F in fuel salt	0.0639	0.1061	0.0860	0.1472	0.3180	0.2297
Core vessel	0.0902	0.1401	0.1093	0.1380	0.1983	0.1508
Li–Be–F in blanket salt	0.0233	0.0234	0.0203	0.0196	0.0215	0.0179
Leakage	0.0477	0.0310	0.0306	0.0193	0.0160	0.0157
Th in fuel salt			0.1095	0.1593		0.1973
Th in blanket salt	0.9722	0.8857	0.8193	0.7066	0.6586	0.5922
Neutron yield, η	2.20	2.19	2.18	2.19	2.21	2.20
Median fission energy, ev	174	14	19	2.9	0.33	1.2
Thermal fissions, %	0.053	8.0	2.3	16	38	29
n–γ capture-to-fission ratio, α	0.14	0.15	0.15	0.15	0.13	0.14
Regeneration ratio	0.97	0.89	0.93	0.87	0.66	0.79

continued

TABLE 14–5 (continued)

Case number	47	48	49	50	51
Fuel and blanket salts*	1	1	1	1	2
Core diameter, ft	8	8	10	10	8
ThF_4 in fuel salt, mole %	0.25	1	0.25	1	0.75
U^{233} in fuel salt, mole %	0.039	0.078	0.031	0.063	0.0597
U^{233} atom density†	1.40	2.95	1.10	2.29	1.97
Critical mass, kg of U^{233}	41.1	86.6	63.0	131	58.8
Critical inventory, kg of U^{233}	93.1	196	104	216	129
Neutron absorption ratios‡					
U^{233} (fissions)	0.8850	0.8755	0.8881	0.8781	0.8809
U^{233} (n–γ)	0.1150	0.1245	0.1119	0.1219	0.1191
Be–Li–F in fuel salt	0.3847	0.1899	0.5037	0.2360	0.2458
Core vessel	0.1406	0.0778	0.1168	0.0629	0.1168
Li–Be–F in blanket salt	0.0141	0.0095	0.0108	0.0071	0.0187
Leakage	0.0095	0.0090	0.0068	0.0065	0.0050
Th in fuel salt	0.2513	0.5768	0.2852	0.6507	0.4903
Th in blanket salt	0.4211	0.3344	0.3058	0.2408	0.3325
Neutron yield, η	2.22	2.20	2.23	2.20	2.21
Median fission energy, ev	0.20	1.1	50% Th	3.2	0.68
Thermal fissions, %	43	24	50	30	34
n–γ capture-to-fission ratio, α	0.13	0.14	0.13	0.14	0.14
Regeneration ratio	0.67	0.91	0.59	0.89	0.82

*Fuel salt No. 1: 31 mole % BeF_2 + 69 mole % LiF + UF_4 + ThF_4
Blanket salt No. 1: 25 mole % ThF_4 + 75 mole % LiF
Fuel salt No. 2: 37 mole % BeF_2 + 63 mole % LiF + UF_4 + ThF_4
Blanket salt No. 2: 13 mole % ThF_4 + 16 mole % BeF_2 + 71 mole % LiF

†Atoms ($\times 10^{-19}$)/cc.

‡Neutrons absorbed per absorption in U^{233}.

TABLE 14-6

NUCLEAR PERFORMANCE OF A TWO-REGION, HOMOGENEOUS, MOLTEN FLUORIDE-SALT REACTOR FUELED WITH U^{233} AND CONTAINING 0.75 MOLE % ThF_4 IN THE FUEL SALT

Core diameter: 8 ft. Total power: 600 Mw (heat).
External fuel volume: 339 ft^3. Load factor: 0.8.

	Initial state			After 1 year		
	Inventory, kg	Absorptions, %	Fissions, %	Inventory, kg	Absorptions, %	Fissions, %
Core elements						
Th^{232}	1,572	22.2		1,572	19.1	
Pa^{233}				9.4	0.5	
U^{233}	129	45.2	100	199	45.3	99.5
U^{234}				23.3	0.9	
U^{235}				1.9	0.3	0.5
U^{236}				0.1	0.1	
Np^{237}						
U^{238}						
Pu^{239}						
Fission fragments				181	7.9	
Li^6	3,920	6.5		3,920	3.4	
Be^9	3,004	0.8		3,008	0.7	
F^{19}	24,000	4.0		24,000	3.5	
Blanket element						
U^{233}				8.6		
Total fuel	129			210		
U^{233} feed rate, kg/day	0.790			0.370–0.189		
Regeneration ratio	0.82			0.71		

continued

TABLE 14-6 (continued)

	After 2 years			After 5 years		
	Inventory, kg	Absorptions, %	Fissions, %	Inventory, kg	Absorptions, %	Fissions, %
Core elements						
Th^{232}	1,572	18.9		1,572	18.3	
Pa^{233}	9.0	0.5		8.9	0.4	
U^{233}	204	44.9	98.5	216	43.7	95.6
U^{234}	44.0	1.7		89	3.1	
U^{235}	5.4	0.8	1.5	17.7	2.3	4.4
U^{236}	0.6	0.3		4.2	0.2	
Np^{237}	0.1	0.1		0.5	0.1	
U^{238}				0.3		
Pu^{239}						
Fission fragments	181	7.7		181	7.2	
Li^{6}	3,920	3.3		3,920	2.8	
Be^{9}	3,008	0.6		3,008	0.6	
F^{19}	24,000	3.4		24,000	3.3	
Blanket element						
U^{233}	10.7			16.2		
Total fuel	220			250		
U^{233} feed rate, kg/day	0.188			0.181		
Regeneration ratio	0.72			0.73		

continued

TABLE 14-6 (continued)

	After 10 years			After 20 years		
	Inventory, kg	Absorptions, %	Fissions, %	Inventory, kg	Absorptions, %	Fissions, %
Core elements						
Th^{232}	1,572	17.8		1,572	17.2	
Pa^{233}	8.6	0.4		8.4	0.4	
U^{233}	231	42.5	92.8	247	41.5	90.5
U^{234}	132	4.2		172	5.0	
U^{235}	32.5	3.7	7.1	47	4.8	9.0
U^{236}	12.5	0.6		24	1.1	
Np^{237}	1.7	0.2		3.4	0.3	
U^{238}	1.7	0.1		5.1	0.3	
Pu^{239}	0.2	0.1	0.1	0.8	0.3	0.5
Fission fragments	181	6.7		181	6.3	
Li^6	3,920	2.5		3,920	2.1	
Be^9	3,008	0.6		3,008	0.6	
F^{19}	24,000	3.3		24,000	3.3	
Blanket element						
U^{233}	22.2			31.6		
Total fuel	282			295		
U^{233} feed rate, kg/day	0.171			0.168		
Regeneration ratio	0.73			0.73		

and Np^{237} constant is begun at this time, the inventory of U^{233} increases slowly to 247 kg and the regeneration ratio rises slightly to 0.73 during the next 19 years. This constitutes a substantial improvement over the performance with U^{235}.

14–3. Homogeneous Reactors Fueled with Plutonium

It may be feasible to burn plutonium in molten fluoride-salt reactors. The solubility of PuF_3 in mixtures of LiF and BeF_2 is considerably less than that of UF_4, but is reported to be over 0.2 mole % [8], which may be sufficient for criticality even in the presence of fission fragments and nonfissionable isotopes of plutonium but probably limits severely the amount of ThF_4 that can be added to the fuel salt. This limitation, coupled with the condition that Pu^{239} is an inferior fuel in intermediate reactors, will result in a poor neutron economy in comparison with that of U^{233}-fueled reactors. However, the advantages of handling plutonium in a fluid fuel system may make the plutonium-fueled molten-salt reactor more desirable than other possible plutonium-burning systems.

14–3.1 Initial states. *Critical concentration, mass, inventory, and regeneration ratio.* The results of calculations of a plutonium-fueled reactor having a core diameter of 8 ft and no thorium in the fuel salt are described below. The critical concentration was 0.013 mole % PuF_3, which is an order of magnitude smaller than the solubility limits in the fluoride salts of interest. The critical mass was 13.7 kg and the critical inventory in a 600-Mw system (339 ft^3 of external fuel volume) was only 31.2 kg.

The core was surrounded by the Li–Be–Th fluoride blanket mixture No. 2 (13% ThF_4). Slightly more than 19% of all neutrons were captured in the thorium to give a regeneration ratio of 0.35. By employing smaller cores and larger investments in Pu^{239}, however, it should be possible to increase the regeneration ratio substantially.

Neutron balance and miscellaneous details. Details of the neutron economy of a reactor fueled with plutonium are given in Table 14–7. Parasitic captures in Pu^{239} are relatively high; η is 1.84, compared with a ν of 2.9. The neutron spectrum is relatively soft; almost 60% of all fissions are caused by thermal neutrons and, as a result, absorptions in lithium are high.

14–3.2 Intermediate states. On the basis of the average value of α of Pu^{239}, it is estimated that Pu^{240} will accumulate in the system until it captures, at equilibrium, about half as many neutrons as Pu^{239}. While these captures are not wholly parasitic, inasmuch as the product, Pu^{241}, is fissionable, the added competition for neutrons will necessitate an increase in the concentration of the Pu^{239}. Likewise, the ingrowth of fission products

will necessitate the addition of more Pu^{239}. Further, the rare earths among the fission products may exert a common-ion influence on the plutonium and reduce its solubility. On the credit side, however, is the U^{233} produced in the blanket. If this is added to the core it may compensate for the ingrowth of Pu^{240} and reduce the Pu^{239} requirement to below the solubility limit, and it may be possible to operate indefinitely, as with the U^{235}-fueled reactors.

14-4. Heterogeneous Graphite-Moderated Reactors

The use of a moderator in a heterogeneous lattice with molten-salt fuels is potentially advantageous. First, the approach to a thermal neutron spectrum improves the neutron yield, η, attainable, especially with U^{235}

Table 14-7

Initial-State Nuclear Characteristics of a Typical Molten Fluoride-Salt Reactor Fueled with Pu^{239}

Core diameter:	8 ft.
External fuel volume:	339 ft^3.
Total power:	600 Mw (heat).
Load factor:	0.8.
Critical inventory:	31.2 kg of Pu^{239}.
Critical concentration:	0.013 mole % Pu^{239}.

	Neutrons absorbed per neutrons absorbed in Pu^{239}
Neutron absorbers	
Pu^{239} (fissions)	0.630
Pu^{239} (n-γ)	0.372
Li^6 and Li^7 in fuel salt	0.202
Be^9 in fuel salt	0.022
F^{19} in fuel salt	0.086
Core vessel	0.145
Th in blanket salt	0.352
Li-Be-F in blanket salt	0.024
Reactor vessel	0.004
Leakage	0.003
Neutron yield, η	1.84
Thermal fissions, %	59
Regeneration ratio	0.352

TABLE 14-8
COMPARISON OF GRAPHITE-MODERATED MOLTEN-SALT AND LIQUID-METAL-FUELED REACTORS

	LMFR	MSFR-1	MSFR-2
Total power, Mw (heat)	580	600	600
Over-all radius, in.	75	75	72
Critical mass, kg of U^{233}	9.9	9.6	27.7
Critical inventory, kg of U^{233}*	467	77.8	213
Regeneration ratio	1.107	0.83	1.07
Core			
Radius, in.	33	33	34.8
Graphite, vol %	45	45	45
Fuel fluid, vol %	55	55	55
Fuel components, mole %			
Bi	~100		
LiF		69	61
BeF_2		31	36.5
ThF_4			2.5
Unmoderated blanket			
Thickness, in.	6	6	13.2
Composition, mole %			
Bi	90		
Th	10 (Th)	10 (ThF_4)	13 (ThF_4)
LiF		70	71
BeF_2		20	16
U^{233}	0.015	0.014	
Moderated blanket			
Thickness, in.	36	36	24
Composition, vol %			
Graphite	66.6	66.6	100
Blanket fluid†	33.4	33.4	
Neutron absorption ratio‡			
Th in fuel fluid			0.566
U^{233} in fuel fluid	0.918	0.925	1.000
Other components of fuel fluid	0.081	0.324	0.106
Th in blanket fluid	1.110	0.825	0.490
U^{233} in blanket fluid	0.083	0.071	
Other components of blanket fluid	0.040	0.092	0.038
Leakage	0.012	0.004	0.014
Neutron yield, η	2.24	2.24	2.21

*With bismuth, the external volume indicated in Ref. 10 was used. The molten-salt systems are calculated for 339 ft^3 external volumes.

†Same as unmoderated blanket fluid.

‡Neutrons absorbed per neutron absorbed in U^{233}.

and Pu^{239}. Second, in a heterogeneous system, the fuel is partially shielded from neutrons of intermediate energy, and a further improvement in effective neutron yield, η, results. Further, the optimum systems may prove to have smaller volumes of fuel in the core than the corresponding fluorine-moderated, homogeneous reactors and, consequently, higher concentrations of fuel and thorium in the melt. This may substantially reduce parasitic losses to components of the carrier salt. On the other hand, these higher concentrations tend to increase the inventory in the circulating-fuel system external to the core. The same considerations apply to fission products and to nonfissionable isotopes of uranium.

Possible moderators for molten-salt reactors include beryllium, BeO, and graphite. The design and performance of the Aircraft Reactor Experiment, a beryllium-oxide moderated, sodium-zirconium fluoride salt, one-region, U^{235}-fueled burner reactor has been reported (see Chapter 16). Since beryllium and BeO and molten salts are not chemically compatible, it was necessary to line the fuel circuit with Inconel. It is easily estimated that the presence of Inconel, or any other prospective containment metal in a heterogeneous thermal reactor would seriously impair the regeneration ratio of a converter-breeder. Consequently, beryllium and BeO are eliminated from consideration.

Preliminary evidence indicates that uranium-bearing molten salts may be compatible with some grades of graphite and that the presence of the graphite will not carburize metallic portions of the fuel circuit seriously [9]. It therefore becomes of interest to explore the capabilities of the graphite-moderated systems. The principal independent variables of interest are the core diameter, fuel channel diameter, lattice spacing, and thorium concentration.

14–4.1 Initial states. Two cases of graphite-moderated molten-salt reactors have been calculated for the same geometry and graphite-to-fluid volume ratio as those for the reference-design LMFR [10]. The results for these two cases, together with those for the liquid bismuth case, are summarized in Table 14–8. Only the initial states are considered, and a metallic shell to separate core and blanket fluids has not been included. With no thorium in the core fluid, the molten-salt-fueled reactor has a significantly lower regeneration ratio than that of the liquid-metal-fueled reactor, with only a slightly lower critical mass. Adding 2.5 mole % ThF_4 to the core fluid increases the initial regeneration ratio to about 1.07, with a critical mass and a corresponding total fuel inventory that are acceptably low.

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