NUCLEAR CHARACTERISTICS OF SPHERICAL,

HOMOGENEOUS, TWO-REGION,

MOLTEN-FLUORIDE-SALT REACTORS

L. G. Alexander
D. A. Carrison
H. G. MacPherson
J. T. Roberts

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

Contract No. W-7405-eng-26

REACTOR PROJECTS DIVISION

NUCLEAR CHARACTERISTICS OF SPHERICAL, HOMOGENEOUS,

TWO-REGION, MOLTEN-FLUORIDE-SALT REACTORS

L. G. Alexander   H. G. MacPherson
D. A. Carrison   J. T. Roberts

DATE ISSUED

SEP 16 1959

CONTENTS

NUCLEAR CHARACTERISTICS OF SPHERICAL, HOMOGENEOUS, TWO-REGION, MOLTEN-FLUORIDE-SALT REACTORS

L. G. Alexander   D. A. Carrison   H. G. MacPherson   J. T. Roberts

ABSTRACT

The use of a molten-salt fuel makes possible the production of high-pressure, superheated steam with a nuclear reactor operating at low pressure. The corrosion resistance of the INOR-8 series of nickel-molybdenum alloys appears to be sufficient to guarantee reactor component life-times of 10 to 20 years. Proposed continuous fuel-processing methods show promise of reducing fuel-processing costs to negligible levels. With U^{233} as the fuel and Th^{232} as the fertile material in both core and blanket, initial regeneration ratios up to 1.08 can be obtained at critical masses less than 600 kg. The corresponding inventory for a 600-Mw(th) central station power reactor is initially about 1300 kg. With U^{235} as the fuel, U^{233} is produced, and initial regeneration ratios in excess of 0.6 can be obtained with critical masses of less than 300 kg. The corresponding critical inventories for 600-Mw(th) central station power reactors are 600 kg or less, depending on the thorium loading. It is concluded that homogeneous, molten-salt-fueled reactors are competitive in regard to nuclear performance with present solid-fuel reactors, and they may be economically superior because of lower fuel and fuel-processing costs.

Molten fluoride salts provide the basis of a new family of liquid-fueled power reactors. The range of solubility of uranium and thorium compounds in the salts makes the system flexible and allows the consideration of a variety of reactors. Suitable salt mixtures have melting points in the range 850 to 950°F and are sufficiently compatible with known alloys to assure long-lived components, if the temperature is kept below 1300°F. As may be seen, molten-salt-fueled reactor systems tend to operate naturally in a temperature region suitable for modern steam plants; they have the further advantage that they achieve these temperatures without pressurization of the molten salt.

The molten-salt system, for purposes other than electric power generation, is not new. Intensive research and development over the past nine years under ORNL sponsorship has provided reasonable answers to a majority of the obvious difficulties. One of the most important advances has been the development of methods for handling salt melts at high temperatures and maintaining them at temperatures above the liquidus temperatures. Information on the chemical and physical properties of a wide variety of molten salts has been obtained, and methods were developed for their manufacture, purification, and handling that are in use on a production scale. It has been found that the simple ionic salts are stable under radiation and that they suffer no deterioration other than the buildup of fission products.

The molten-salt system has the usual benefits attributed to fluid-fueled systems. The principal advantages claimed over solid-fuel systems are: (1) the lack of radiation damage that can limit fuel burnup; (2) the avoidance of the expense of fabricating new fuel elements; (3) the continuous removal of fission products; (4) a high negative temperature coefficient of reactivity; and (5) the avoidance of the need for excess reactivity, since makeup fuel can be added as required. The last two factors make possible a reactor without control rods that automatically adjusts its power in response to changes of the electrical load. The lack of excess reactivity can lead to a reactor that is safe from nuclear power excursions.

In comparison with aqueous systems, the molten-salt system has three outstanding advantages: it allows high-temperature operation at a low pressure; explosive radiolytic gases are not formed; and thorium compounds are soluble in the salts. Compensating disadvantages are the high melting

points of the salts and poorer neutron economy; the importance of these disadvantages cannot be assessed properly without further experience.

Probably the most outstanding characteristic of the molten-salt systems is their chemical flexibility, that is, the wide variety of molten-salt solutions which are available for reactor use. In this respect, the molten-salt systems are practically unique, and this is the essential advantage which they enjoy over the uranium-bismuth systems. Thus the molten-salt systems are not to be thought of in terms of a single reactor — rather, they are the basis for a new class of reactors. Included in this class are all the embodiments which comprise the whole of solid-fuel-element technology: U^{235} burners, thorium-uranium thermal converters or breeders, and thorium-uranium fast converters or breeders. Of possible short-term interest is the U^{235} burner. Because of the inherently high temperatures and because there are no fuel elements, the fuel cost in the salt system is of the order of 1 mill/kwhr in graphite-moderated, molten-salt-fueled reactors.

The present technology suggests that homogeneous converters using a base salt composed of BeF_2 and either Li^7F or NaF and using UF_4 for fuel and ThF_4 for a fertile material are more suitable for early reactors than are graphite-moderated or plutonium-fueled systems. The chief virtues of the homogeneous converter reactor are that it is based on well-explored principles and that the use of a simple fuel cycle should lead to low fuel cycle costs.

With further development, the same base salt (using Li^7F) can be combined with a graphite moderator in a heterogeneous arrangement to provide a self-sustaining Th-U^{233} system with a breeding ratio of about 1. The chief advantage of the molten-salt system over other liquid systems in pursuing this objective is, as has been mentioned, that it is the only system in which a soluble thorium compound can be used, and thus the problem of slurry handling is avoided.

PRIOR WORK

The applicability of molten salts to nuclear reactors has been ably discussed by Grimes and others.[1,2] The most promising systems are those comprising the fluorides and chlorides of the alkali metals, zirconium, and beryllium. These appear to possess the most desirable combination of low neutron absorption, high solubility of uranium and thorium, and chemical inertness. In general, the chlorides have lower melting points, but they appear to be less stable and more corrosive than the fluorides.

The fluoride systems appear to be preferable for use in thermal and epithermal reactors. Many mixtures have been investigated, mainly at ORNL and at Mound Laboratory. The physical properties of these mixtures, in so far as they are known, have been tabulated, and the results of extensive phase studies have been reported.[3]

Lithium-7 has an attractively low capture cross section, 0.0189 barn at 0.0759 ev, but Li^6, which comprises 7.5% of the natural mixture, has a capture cross section of 542 barns at the same energy. The cross sections at 0.0759 ev and 1150°F for several lithium compositions are compared below with the cross sections of sodium, potassium, rubidium, and cesium.

Element	Cross Section (barns)
Lithium	
0.1% Li^6	0.561
0.01% Li^6	0.0731
0.001% Li^6	0.0243
0.0001% Li^6	0.0194
Sodium	0.290
Potassium	1.130
Rubidium	0.401
Cesium	29

The capture cross sections of the lighter elements at higher energies presumably stand in approximately the same relation as at thermal. It may be seen that purified Li^7 has an attractively low cross section in comparison with the cross sections of other alkali metals and that sodium is the next best alkali metal.

The sodium-zirconium fluoride system has been extensively studied at ORNL.[3] A eutectic containing about 42 mole % ZrF_4 melts at 910°F.

[1] W. R. Grimes, ORNL CF-52-4-197, p 320 ff (April 1952) (classified).

[2] W. R. Grimes, D. R. Cuneo, and F. F. Blankenship, in *Reactor Handbook*, ed. by J. F. Hogerton and R. C. Grass, vol 2, sec 6, p 799, AECD-3646 (May 1955).

[3] J. A. Lane, H. G. MacPherson, and F. Maslan (eds.), *Fluid Fuel Reactors*, p 569, Addison-Wesley, Reading, Mass., 1958.

Small additions of UF_4 lower the melting point appreciably. A fuel of this type was successfully used in the Aircraft Reactor Experiment (ARE).[4] Inconel, a nickel-rich alloy, is reasonably resistant to corrosion by this fuel system at 1500°F. Although long-term data are lacking, there is reason to expect the corrosion rate at 1200°F to be sufficiently low that Inconel equipment would last several years.

However, with regard to its use in a central-station power reactor, the sodium-zirconium fluoride system has several serious disadvantages. The sodium capture cross section is less favorable than the Li^7 cross section. In addition, there is the so-called "snow" problem; that is, ZrF_4 tends to evaporate from the fuel and crystallize on surfaces exposed to the vapor. In comparison with the lithium-beryllium system discussed below, the sodium-zirconium system has inferior heat transfer and cooling effectiveness. Finally, the expectation at Oak Ridge is that the INOR-8 alloys will prove to be as resistant to the beryllium salts as to the zirconium salts and that there is therefore no compelling reason for selecting the sodium-zirconium system.

The capture cross section of beryllium appears to be satisfactorily low at all energies. A phase diagram for the system $LiF-BeF_2$ has recently been published.[5] A mixture containing 31 mole % BeF_2 reportedly liquefies at approximately 980°F. Substantial concentrations of ThF_4 in the core fluid may be obtained by blending this mixture with the compound $3LiF \cdot ThF_4$. A temperature diagram for the ternary system has been published.[6] The *liquidus temperature along the join appears to lie below 930°F for mixtures containing up to 10 mole % ThF_4.* Small additions of UF_4 to any of these mixtures should lower the liquidus temperature somewhat.

The ARE was operated with a molten-fluoride-salt fuel in November 1954. The reactor had a moderator consisting of beryllium oxide blocks. The fuel, which was a mixture of sodium fluoride, zirconium fluoride, and uranium fluoride, flowed through the moderator in Inconel tubes and was pumped through an external heat exchanger by

means of a centrifugal pump. The reactor operated at a peak power of 2.5 Mw. It was dismantled after carrying out a scheduled experimental program.

In 1953 a group of ORSORT students, under the leadership of Jarvis,[7] investigated the applicability of molten salts to package reactors. More recently, another ORSORT group led by Davies prepared a valuable study of the feasibility of molten-salt U^{235} burners for central-station power production.[8] Fast reactors based on the U^{238}-Pu cycle were studied by Addoms et al. of MIT and by an ORSORT group led by Bulmer.[9] Both groups concluded that it would be preferable to use molten chlorides, rather than the fluorides, because of the relatively high moderating power of the fluorine nucleus, although it was recognized that the chlorides are probably inferior with respect to corrosion and radiation stability. Bulmer et al. also pointed out that it would be necessary to use purified Cl^{37} on account of the (n,p) reaction exhibited by Cl^{35}. Because of the disadvantages of the chloride systems and, further, because the technology of handling and utilizing neptunium- and plutonium-bearing salts is largely unknown, it was decided to postpone consideration of chloride salt reactors.

In 1953 an ORSORT group led by Wehmeyer[10] analyzed many of the problems presently under study. The proposals set forth in that report have influenced the present program. A study by Davidson and Robb of KAPL[11] has also been helpful. Both studies considered the possibility of using thorium in a U^{233} conversion-breeding cycle at thermal or near thermal energies.

A recent conceptual design study[12] of a 240-Mw (electrical) central-station molten-salt-fueled reactor was used as a basis for examining the economics and feasibility of a reactor using molten-salt fuel. An attempt was made to keep the

[7] T. Jarvis et al., ORNL CF-53-10-26 (August 1953) (classified).

[8] R. W. Davies et al., ORNL CF-56-8-208 (August 1956) (classified).

[9] J. Bulmer et al., Fused Salt Fast Breeder, ORNL CF-56-8-204 (August 1956).

[10] D. B. Wehmeyer et al., Study of a Fused Salt Breeder Reactor for Power Production, ORNL CF-53-10-25 (September 1953).

[11] J. K. Davidson and W. L. Robb, A Molten-Salt Thorium Converter for Power Production, KAPL-M-JKD-10 (Oct. 15, 1956).

[12] Lane, MacPherson, and Maslan, op. cit., p 681.

[4] Ibid., p 673.

[5] Ibid., p 573.

[6] Ibid., p 579.

technology and the processing scheme as simple as possible.

METHOD OF CALCULATION

Reactor calculations were performed by means of the UNIVAC program Ocusol,[13] a modification of the Eyewash program.[14] 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.[15] Where data were lacking, reasonable interpolations based on resonance theory were used. The estimated cross sections were made to agree with measured resonance integrals where available. Saturations and Doppler broadening of the resonances in thorium as a function of concentration were estimated.

The molten salts may be used as homogeneous moderators or simply as fuel carriers in heterogeneous reactors. Although 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 prior 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, spherical regions: (1) the core; (2) an INOR-8 reactor vessel, $\frac{1}{3}$ in. thick; (3) a blanket, approximately 2 ft thick; and (4) an INOR-8 reactor vessel, $\frac{2}{3}$ 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., could be readily calculated.

HOMOGENEOUS REACTORS FUELED WITH U^{235}

While the isotope U^{233} would be a superior fuel in molten-fluoride-salt reactors, 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 the first phase are given here. Briefly, it was found that regeneration ratios of up to 0.65 could be obtained with moderate investment in U^{235} (less than 100 kg).

Initial States

A complete parametric study was made 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. 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/cm^3 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/cm^3 at 1150°F. In order to shorten the calculations in this series, the reactor vessel was neglected, since the resultant error would be 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 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, that is, the number of neutrons produced by fission per neutron absorbed in fuel. Further, the sum of the absorptions in U^{238} and thorium in the fuel and in thorium in the blanket

[13]L. G. Alexander *et al.*, *Operating Instructions for the Univac Program Ocusol-A, a Modification of the Eyewash Program*, ORNL CF-57-6-4 (June 5, 1957).

[14]J. H. Alexander and N. D. Given, *A Machine Multigroup Calculation. The Eyewash Program for Univac*, ORNL-1925 (Sept. 15, 1955).

[15]J. T. Roberts and L. G. Alexander, *Cross Sections for Ocusol-A Program*, ORNL CF-57-6-5 (June 11, 1957).

Table 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	8	9	10	11
Core diameter, ft	4	5	5	5	5	5	6	6	6	6	6
ThF_4 in fuel salt, mole %	0	0	0.25	0.5	0.75	1	0	0.25	0.5	0.75	1
U^{235} in fuel salt, mole %	0.952	0.318	0.561	0.721	0.845	0.938	0.107	0.229	0.408	0.552	0.662
U^{235} atom density*	33.8	11.3	20.1	25.6	30.0	33.3	3.80	8.13	14.5	19.6	23.5
Critical mass, kg of U^{235}	124	81.0	144	183	215	239	47.0	101	179	243	291
Neutron absorption ratios**											
U^{235} (fissions)	0.7023	0.7185	0.7004	0.6996	0.7015	0.7041	0.7771	0.7343	0.7082	0.7000	0.7004
U^{235} (n,γ)	0.2977	0.2815	0.2996	0.3004	0.2985	0.2959	0.2229	0.2657	0.2918	0.3000	0.2996
Be, Li, and F in fuel salt	0.0551	0.0871	0.0657	0.0604	0.0581	0.0568	0.1981	0.1082	0.0770	0.0669	0.0631
Core vessel	0.0560	0.0848	0.0577	0.0485	0.0436	0.0402	0.1353	0.0795	0.0542	0.0435	0.0388
Li and F in blanket salt	0.0128	0.0138	0.0108	0.0098	0.0093	0.0090	0.0164	0.0116	0.0091	0.0081	0.0074
Leakage	0.0229	0.0156	0.0147	0.0143	0.0141	0.0140	0.0137	0.0129	0.0122	0.0119	0.0116
U^{238} in fuel salt	0.0430	0.0426	0.0463	0.0451	0.0431	0.0412	0.0245	0.0375	0.0477	0.0467	0.0452
Th in fuel salt			0.0832	0.1289	0.1614	0.1873		0.1321	0.1841	0.2142	0.2438
Th in blanket salt	0.5448	0.5309	0.4516	0.4211	0.4031	0.3905	0.5312	0.4318	0.3683	0.3378	0.3202
Neutron yield, η	1.73	1.77	1.73	1.73	1.73	1.74	1.92	1.82	1.75	1.73	1.73
Median fission energy, ev	270	15.7	105	158	270	425	0.18	5.6	38	100	120
Thermal fissions, %	0.052	6.2	0.87	0.22	0.87	0.040	35	13	3	0.56	0.48
Regeneration ratio	0.59	0.57	0.58	0.60	0.61	0.62	0.56	0.61	0.60	0.60	0.61

*10^{19} atoms/cm^3.
**Neutrons absorbed per neutron absorbed in U^{235}.

Table 1 (continued)

Case number	12	13	14	15	16	17	18	19	20	21	22
Core diameter, ft	7	8	8	8	8	8	10	10	10	10	10
ThF_4 in fuel salt, mole %	0.25	0	0.25	0.5	0.75	1	0	0.25	0.5	0.75	1
U^{235} in fuel salt, mole %	0.114	0.047	0.078	0.132	0.226	0.349	0.033	0.052	0.081	0.127	0.205
U^{235} atom density*	4.05	1.66	2.77	4.67	8.03	12.4	1.175	1.86	2.88	4.50	7.28
Critical mass, kg of U^{235}	79.6	48.7	81.3	137	236	364	67.3	107	165	258	417
Neutron absorption ratios**											
U^{235} (fission)	0.7748	0.8007	0.7930	0.7671	0.7362	0.7146	0.8229	0.7428	0.7902	0.7693	0.7428
U^{235} (n,γ)	0.2252	0.1993	0.2070	0.2329	0.2638	0.2854	0.1771	0.2572	0.2098	0.2307	0.2572
Be, Li, and F in fuel salt	0.1880	0.4130	0.2616	0.1682	0.1107	0.0846	0.5713	0.3726	0.2486	0.1735	0.1206
Core vessel	0.0951	0.1491	0.1032	0.0722	0.0500	0.0373	0.1291	0.0915	0.0669	0.0497	0.0363
Li and F in blanket salt	0.0123	0.0143	0.0112	0.0089	0.0071	0.0057	0.0114	0.0089	0.0073	0.0060	0.0049
Leakage	0.0068	0.0084	0.0082	0.0080	0.0077	0.0074	0.0061	0.0060	0.0059	0.0057	0.0055
U^{238} in fuel salt	0.0254	0.0143	0.0196	0.0272	0.0368	0.0428	0.0120	0.0153	0.0209	0.0266	0.0343
Th in fuel salt	0.1761		0.2045	0.3048	0.3397	0.3515		0.2409	0.3691	0.4324	0.4506
Th in blanket salt	0.4098	0.4073	0.3503	0.3056	0.2664	0.2356	0.3031	0.2617	0.2332	0.2063	0.1825
Neutron yield, η	191	2.00	1.96	1.89	1.82	1.76	2.03	2.00	1.95	1.90	1.83
Median fission energy, ev	0.16	Thermal	0.10	0.17	5.3	27	Thermal	Thermal	0.100	0.156	1.36
Thermal fissions, %	33	59	45	29	13	5	66	56	43	30	16
Regeneration ratio	0.61	0.42	0.57	0.64	0.64	0.63	0.32	0.52	0.62	0.67	0.67

*10^{19} atoms/cm^3.

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

salt give directly the regeneration ratio. The losses to other elements are penalties imposed on the regeneration ratio by these poisons.

A graph of critical mass plotted as a function of core diameter, with thorium concentration as a parameter, is presented in Fig. 1. The masses vary from about 40 kg of U^{235} in a 7-ft-dia core having no thorium in the fuel to about 450 kg in the 10-ft-dia core having 1 mole % ThF_4 in the
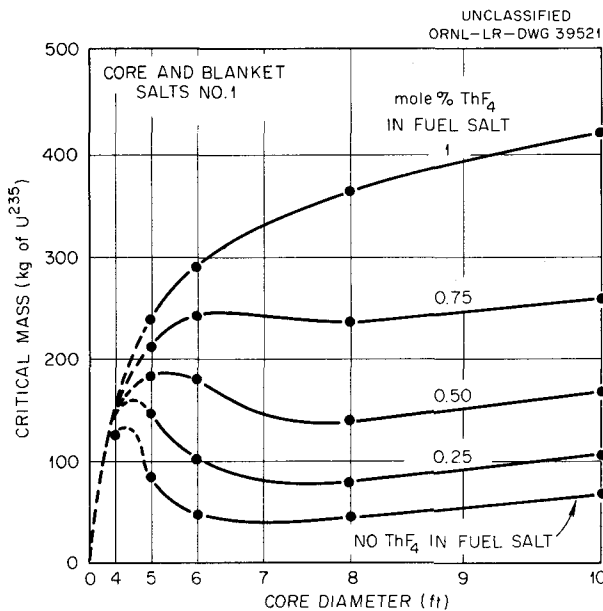

Fig. 1. Initial Critical Masses of U^{235} in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors.

fuel. The corresponding regeneration ratios, plotted in Fig. 2, range from 0.5 for the minimum mass reactor to 0.63 for the largest mass reactor. It does not seem likely that further increases in diameter or thorium concentration would increase the regeneration above 0.65.

The effects of changes in the compositions of the fuel and blanket salts were studied in a series of calculations for salts having more favorable melting points and viscosities. The BeF_2 content was raised to 37 mole % in the fuel salt (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 is a somewhat better moderator than No. 1. As a result, at a given core diameter and thorium
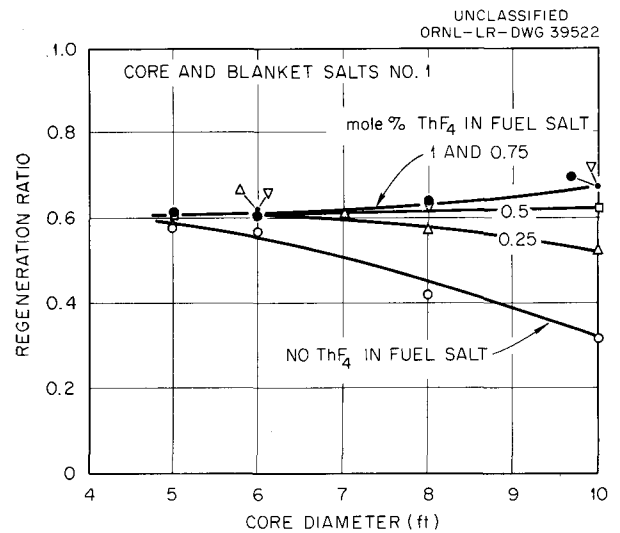

Fig. 2. Initial Fuel Regeneration in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{235}.

concentration in the fuel salt, both the critical concentration and the regeneration ratio were 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 of the No. 2 fuel and blanket salts in reactors with core diameters in the 6- to 8-ft range was made. In this study the presence of an outer reactor vessel consisting of $\frac{2}{3}$ in. of INOR-8 was taken into account. The results are presented in Table 2. In general, the nuclear performance is somewhat better with the No. 2 salts than with the No. 1 salts.

Neutron Balances and Other Reactor Variables

The distributions of the neutron captures are given in Tables 1 and 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 lithium, beryllium, and fluorine 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

8

Table 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	29	30	31	32	33	34
Core diameter, ft	6	6	6	6	7	7	7	7	8	8	8	8
ThF_4 in fuel salt, mole %	0.25	0.5	0.75	1	0.25	0.5	0.75	1	0.25	0.5	0.75	1
U^{235} in fuel salt, mole %	0.169	0.310	0.423	0.580	0.084	0.155	0.254	0.366	0.064	0.099	0.163	0.254
U^{235} atom density*	5.87	10.91	15.95	20.49	3.13	5.38	8.70	13.79	2.24	3.51	5.62	9.09
Critical mass, kg of U^{235}	72.7	135	198	254	61.5	106	171	271	65.7	103	165	267
Neutron absorption ratios**												
U^{235} (fissions)	0.7516	0.7174	0.7044	0.6958	0.7888	0.7572	0.7282	0.7094	0.8014	0.7814	0.7536	0.7288
U^{235} (n,γ)	0.2484	0.2826	0.2956	0.3042	0.2112	0.2428	0.2718	0.2906	0.1986	0.2186	0.2464	0.2712
Be, Li, and F in fuel salt	0.1307	0.0900	0.0763	0.0692	0.2147	0.1397	0.1010	0.0824	0.2769	0.1945	0.1354	0.1016
Core vessel	0.1098	0.0726	0.0575	0.0473	0.1328	0.0905	0.0644	0.0497	0.1308	0.0967	0.0696	0.0518
Li and F in blanket salt	0.0214	0.0159	0.0132	0.0117	0.0215	0.0167	0.0131	0.0108	0.0198	0.0162	0.0130	0.0105
Outer vessel	0.0024	0.0021	0.0021	0.0019	0.0019	0.0018	0.0016	0.0015	0.0017	0.0016	0.0014	0.0013
Leakage	0.0070	0.0065	0.0064	0.0061	0.0052	0.0050	0.0048	0.0045	0.0045	0.0043	0.0042	0.0040
U^{238} in fuel salt	0.0325	0.0426	0.0452	0.0477	0.0214	0.0307	0.0392	0.0447	0.0177	0.0233	0.0315	0.0392
Th in fuel salt	0.1360	0.1902	0.2212	0.2387	0.1739	0.2565	0.2880	0.3022	0.1978	0.3043	0.3501	0.3637
Th in blanket salt	0.4165	0.3521	0.3178	0.2962	0.3770	0.3294	0.2866	0.2566	0.3240	0.2892	0.2561	0.2280
Neutron yield, η	1.86	1.77	1.74	1.72	1.95	1.87	1.80	1.75	1.97	1.93	1.86	1.80
Median fission energy, ev	0.480	10.47	58.10	76.1	0.1223	0.415	7.61	25.65	51% th	0.136	0.518	7.75
Thermal fissions, %	21	7	2.8	0.84	43	24	11	4.3	51	38	23	11
Regeneration ratio	0.59	0.58	0.58	0.58	0.57	0.62	0.61	0.60	0.54	0.62	0.64	0.63

*10^{19} atoms/cm^3.

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

the hardness decreases losses to salt and core vessel sharply (case No. 5) but increases the loss to the (n,γ) reaction. The numbers given for capture in the lithium and fluorine 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-dia core containing 1 mole % ThF_4.

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

The mean neutron energy is somewhat nearer thermal in such 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 are reduced substantially at higher thorium concentrations because of the hardening of the neutron spectrum.

Results from a parametric study of the nuclear characteristics of two-region, homogeneous, molten-fluoride-salt reactors fueled with U^{233} are given in Tables 3 and 4. The core diameters considered range from 3 to 12 ft, and the thorium concentrations range from 0.25 to 7 mole %. The regeneration ratios are very good compared with those obtained with U^{235}. With 7 mole % ThF_4 in an 8-ft-dia core, the U^{233} critical mass was 1500 kg, and the regeneration ratio was 1.09.

The data in Table 3 are for reactors using fuel and blanket salts No. 1 with ThF_4 concentrations ranging up to 1 mole %. The critical masses are graphed in Fig. 3 and the regeneration ratios in Fig. 4. The masses range from a minimum of about

20 kg in a 5-ft-dia core, with no thorium present, to 130 kg in a 10-ft-dia core having 1 mole % thorium in the fuel. The corresponding regeneration ratios are 0.60 and 0.90. For a given thorium
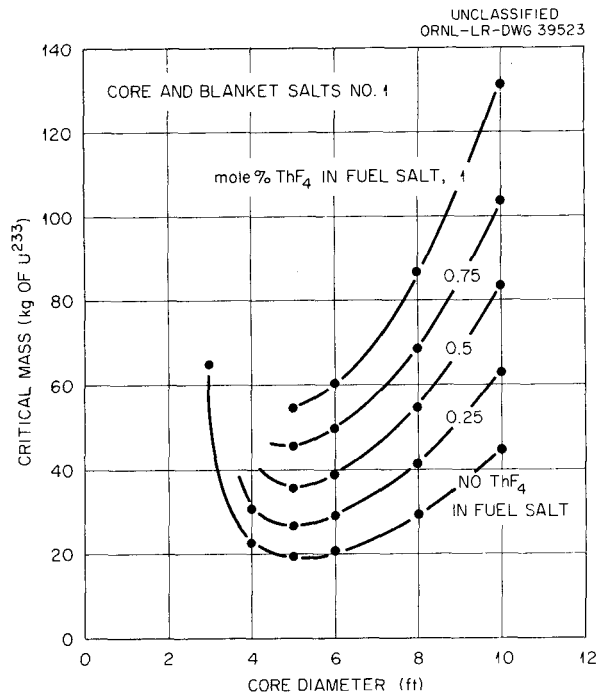

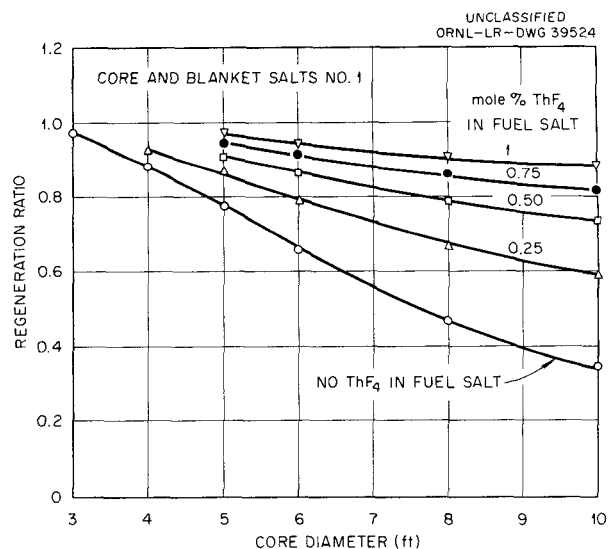Fig. 3. Critical Masses of Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}.

Fig. 4. Initial Fuel Regeneration in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}.

9

Table 3. Initial-State Nuclear Characteristics of Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}

Fuel salt No. 1: 31 mole % BeF_2 + 69 mole % LiF + UF_4 + ThF_4

Blanket salt: 25 mole % ThF_4 + 75 mole % LiF

Total power: 600 Mw (heat)

External fuel volume: 339 ft^3

Case number	35	36	37	38	39	40	41	42	43	44	45	46
Core diameter, ft	3	4	4	5	5	5	5	5	6	6	6	6
ThF_4 in fuel salt, mole %	0	0	0.25	0	0.25	0.5	0.75	1	0	0.25	0.5	0.75
U^{233} in fuel salt, mole %	0.592	0.158	0.233	0.076	0.106	0.141	0.179	0.214	0.048	0.066	0.087	0.113
U^{233} atom density*	21.1	5.6	8.26	2.7	3.73	5.0	6.35	7.605	1.7	2.3	3.1	4.0
Critical mass, kg of U^{233}	64.9	22.3	30.3	19.3	26.9	35.8	45.5	54.5	20.4	29.2	38.4	49.5
Neutron absorption ratios**												
U^{233} (fissions)	0.8754	0.8706	0.8665	0.8767	0.8725	0.8684	0.8674	0.8672	0.8814	0.8779	0.8744	0.8665
U^{233} (n,γ)	0.1246	0.1294	0.1335	0.1233	0.1275	0.1316	0.1326	0.1328	0.1186	0.1221	0.1256	0.1335
Be, Li, and F in fuel salt	0.0639	0.1051	0.0860	0.1994	0.1472	0.1174	0.1010	0.0905	0.3180	0.2297	0.1774	0.1412
Core vessel	0.0902	0.1401	0.1093	0.1808	0.1380	0.1112	0.0944	0.0821	0.1983	0.1508	0.1209	0.0989
Li and F in blanket salt	0.0233	0.0234	0.0203	0.0232	0.0196	0.0172	0.0157	0.0146	0.0215	0.0179	0.0157	0.0139
Leakage	0.0477	0.0310	0.0306	0.0197	0.0193	0.0190	0.0189	0.0188	0.0160	0.0157	0.0157	0.0154
Th in fuel salt	0.0000	0.0000	0.1095	0.0000	0.1593	0.2561	0.3219	0.3702	0.0000	0.1973	0.3111	0.3989
Th in blanket salt	0.9722	0.8857	0.8193	0.7777	0.7066	0.5487	0.6255	0.6004	0.6586	0.5922	0.5539	0.5169
Neutron yield, η	2.1973	2.1853	2.1750	2.2007	2.1900	2.1797	2.1773	2.1766	2.2124	2.2035	2.1948	2.185
Median fission energy, ev	174	14.2	19.1	1.752	2.87	9.625	16.5	29.35	0.326	1.18	2.175	10.16
Thermal fissions, %	0.0527	7.952	2.970	24.80	16.499	10.09	5.99	3.192	37.832	29.37	27.12	14.87
Regeneration ratio	0.9722	0.8856	0.9288	0.7777	0.8659	0.9148	0.9474	0.9706	0.5486	0.7895	0.8651	0.9158

*10^{19} atoms/cm^3.

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

Table 3 (continued)

Case number	47	48	49	50	51	52	53	54	55	56	57
Core diameter, ft	6	8	8	8	8	8	10	10	10	10	10
ThF_4 in fuel salt, mole %	1	0	0.25	0.5	0.75	1	0	0.25	0.5	0.75	1
U^{233} in fuel salt, mole %	0.133	0.028	0.039	0.052	0.066	0.078	0.022	0.031	0.041	0.051	0.063
U^{233} atom density*	4.72	1.01	1.41	1.85	2.33	2.72	0.780	1.09	1.45	1.8	2.25
Critical mass, kg of U^{233}	58.4	29.6	41.1	54.3	68.4	86.6	44.7	63.0	83.1	103.2	131.3
Neutron absorption ratios**											
U^{233} (fissions)	0.8693	0.8876	0.8850	0.8808	0.8779	0.8755	0.8921	0.8881	0.8842	0.8814	0.8781
U^{233} (n,γ)	0.1307	0.1124	0.1150	0.1192	0.1221	0.1245	0.1079	0.1119	0.1158	0.1186	0.1219
Be, Li, and F in fuel salt	0.1216	0.5433	0.3847	0.2896	0.2285	0.1899	0.7166	0.5037	0.3758	0.2952	0.2360
Core vessel	0.0855	0.1866	0.1406	0.1112	0.0915	0.0778	0.1560	0.1168	0.0919	0.0754	0.0629
Li and F in blanket salt	0.0127	0.0176	0.0141	0.0120	0.0106	0.0095	0.0133	0.0108	0.0091	0.0080	0.0071
Leakage	0.0152	0.0095	0.0095	0.0093	0.0091	0.0090	0.0068	0.0068	0.0066	0.0065	0.0065
Th in fuel salt	0.4580	0.0000	0.2513	0.4044	0.5055	0.5768	0.0000	0.2852	0.4585	0.5708	0.6507
Th in blanket salt	0.4889	0.4707	0.4211	0.3842	0.3582	0.3344	0.3466	0.3058	0.2774	0.2564	0.2408
Neutron yield, η	2.1820	2.2277	2.2212	2.2108	2.2035	2.1975	2.2392	2.2290	2.2194	2.2133	2.2040
Median fission energy, ev	8.51	52% th	0.197	0.4915	1.185	1.12	58% th	50% th	0.1735	0.455	3.25
Thermal fissions, %	12.42	51.93	43.398	35.79	29.078	24.36	58.34	50.39	42.8	36.45	29.96
Regeneration ratio	0.9470	0.4707	0.6725	0.7886	0.8638	0.9112	0.3467	0.5910	0.7359	0.8271	0.8915

*10^{19} atoms/cm^3.

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

Table 4. Initial-State Nuclear Characteristics of Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}

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	58	59	60	61	62	63	64	65	66	67	68	69	70
Core diameter, ft	6	6	6	6	7	7	7	7	8	8	8	8	4
Th_4 in fuel salt, mole %	0.25	0.5	0.75	1	0.25	0.5	0.75	1	0.25	0.5	0.75	1	2
U^{233} in fuel salt, mole %	0.062	0.081	0.10	0.121	0.047	0.059	0.074	0.091	0.039	0.049	0.062	0.075	0.619
U^{233} atom density*	1.98	2.6	3.2	3.88	1.5	1.92	2.38	2.9	1.21	1.58	1.97	2.41	19.8
Critical mass, kg of U^{233}	24.51	32.19	39.62	48.03	29.49	37.75	46.79	57.01	35.51	46.37	57.82	70.73	72.0
Neutron absorption ratios**													
U^{233} (fissions)	0.8805	0.8762	0.8741	0.8722	0.8843	0.8809	0.8784	0.8749	0.8880	0.8828	0.8809	0.8827	0.871
U^{233} (n,γ)	0.1195	0.1238	0.1259	0.1278	0.1157	0.1191	0.1216	0.1251	0.1120	0.1172	0.1191	0.1173	0.129
Be, Li, and F in fuel salt	0.2427	0.1915	0.1604	0.1383	0.3209	0.2525	0.2062	0.1735	0.2407	0.3051	0.2458	0.2073	0.070
Core vessel	0.1891	0.1526	0.1288	0.1109	0.1858	0.1505	0.1258	0.1070	0.1756	0.1405	0.1168	0.1003	0.073
Li and F in blanket salt	0.0313	0.0272	0.0243	0.0221	0.0276	0.0238	0.0211	0.0190	0.0247	0.0212	0.0187	0.0169	0.025
Leakage	0.0133	0.0111	0.0109	0.0108	0.0094	0.0081	0.0080	0.0078	0.0070	0.0069	0.0068	0.0068	0.031
Th in fuel salt	0.1901	0.3088	0.3902	0.4504	0.2182	0.3531	0.4455	0.5125	0.3952	0.3891	0.4903	0.5678	0.343
Th in blanket salt	0.5454	0.5079	0.4794	0.4566	0.4589	0.4228	0.3983	0.3763	0.3859	0.3533	0.3325	0.3164	0.653
Neutron yield, η	2.2100	2.1992	2.1940	2.1891	2.2197	2.2110	2.2049	2.1960	2.2289	2.2160	2.2110	2.2155	2.195
Median fission energy, ev	0.721	1.575	2.475	3.685	0.1875	0.465	0.992	2.025	0.1223	0.230	0.676	1.345	147
Thermal fissions, %	33.878	26.269	20.518	15.584	41.997	35.191	28.685	23.051	47.965	40.663	33.87	28.301	0.23
Regeneration ratio	0.7355	0.8167	0.8695	0.9071	0.6770	0.7760	0.8438	0.8887	0.6264	0.7424	0.8228	0.8842	0.996

*10^{19} atoms/cm^3.

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

Table 4 (continued)

Case number	71	72	73	74	75	76	77	78	79	80	81	82	83	84
Core diameter, ft	4	4	6	6	6	8	8	8	10	10	10	12	12	12
ThF$_4$ in fuel salt, mole %	4	7	2	4	7	2	4	7	2	4	7	2	4	7
U^{233} in fuel salt, mole %	0.856	1.247	0.236	0.450	0.762	0.152	0.316	0.603	0.121	0.262	0.528	0.101	0.222	0.477
U^{233} atom density*	27.4	39.9	7.55	14.4	24.4	4.88	10.1	19.3	3.86	8.39	16.9	3.24	7.39	15.25
Critical mass, kg of U^{233}	100.5	146.5	94.2	177.8	301	143	299	566	221	481	970	320	732	1510
Neutron absorption ratios**														
U^{233} (fissions)	0.874	0.881	0.864	0.868	0.876	0.867	0.865	0.873	0.870	0.864	0.871	0.873	0.864	0.870
U^{233} (n,γ)	0.126	0.119	0.136	0.132	0.124	0.133	0.135	0.127	0.130	0.136	0.129	0.127	0.136	0.130
Be, Li, and F in fuel salt	0.066	0.069	0.093	0.075	0.076	0.120	0.082	0.078	0.142	0.088	0.081	0.164	0.093	0.083
Core vessel	0.059	0.048	0.068	0.049	0.035	0.057	0.037	0.025	0.046	0.030	0.018	0.033	0.022	0.012
Li and F in blanket salt	0.021	0.019	0.016	0.014	0.011	0.019	0.012	0.009	0.009	0.006	0.006	0.012	0.004	0.004
Leakage	0.031	0.028	0.017	0.017	0.015	0.010	0.010	0.009	0.007	0.008	0.007	0.004	0.006	0.004
Th in fuel salt	0.426	0.517	0.581	0.650	0.740	0.716	0.785	0.865	0.800	0.865	0.938	0.872	0.922	0.998
Th in blanket salt	0.600	0.538	0.403	0.382	0.330	0.264	0.254	0.2113	0.189	0.170	0.146	0.115	0.130	0.092
Neutron yield, η	2.203	2.219	2.1770	2.187	2.2072	2.1860	2.180	2.200	2.1933	2.177	2.196	2.1995	2.177	2.1931
Median fission energy, ev	503	1085	24.2	64.0	443	8.42	51.3	243	3.64	45.9	193	2.45	41.4	178
Thermal fissions, %	0.11	0.084	4.3	0.35	0.076	11.0	1.5	0.091	17	1.8	0.12	21	2.1	0.15
Regeneration ratio	1.026	1.055	0.984	1.032	1.070	0.980	1.039	1.078	0.989	1.045	1.084	0.987	1.052	1.090

*10^{19} atoms/cm^3

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

concentration, the regeneration ratio tends to increase with decreasing core size, and ratios up to 0.97 were observed in this series of calculations, as shown in Fig. 4.

The data in Table 4 are for reactors using fuel and blanket salts No. 2. In this series of calculations, the diameter ranged up to 12 ft and the thorium concentration in the core up to 7 mole %. It was necessary to alter progressively the composition of the base salt as the thorium concentration was increased in order to keep the liquidus temperature below 1000°F. There was a slight increase in concentration of LiF at the expense of BeF_2. For cores having thorium concentrations in the range from 0.25 to 1 mole %, the results are about the same as those obtained with cores using No. 1 salts. The behavior with No. 2 salts at higher concentrations of ThF_4 is shown in Figs. 5 and 6. It is seen that an initial regeneration ratio of about 1.0 can be achieved with about 2.5 mole % thorium in the fuel, regardless of the diameter of
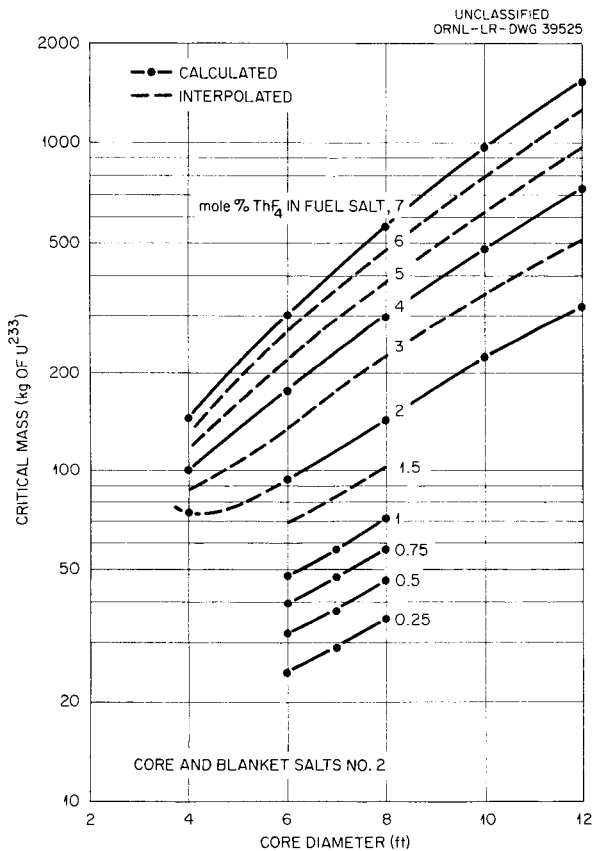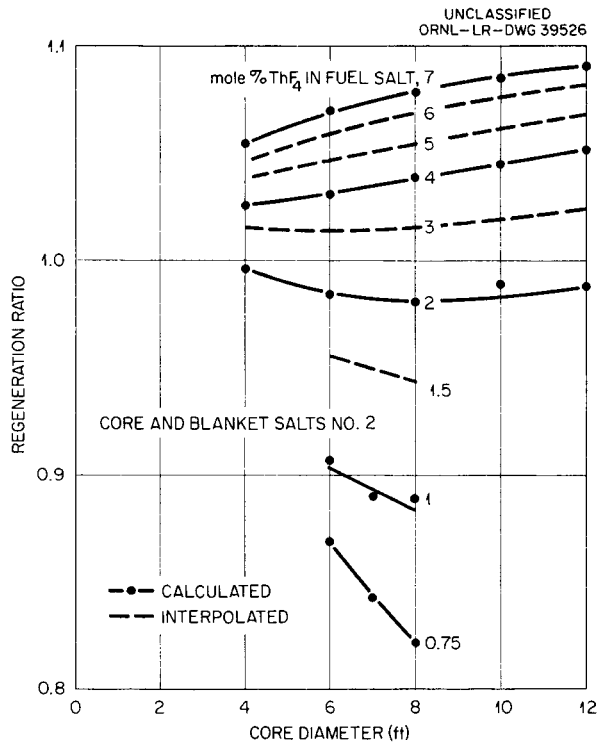
Fig. 6. Initial Regeneration Ratio in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}.

the core in the range from 4 to 12 ft. The corresponding critical masses range from about 80 to 400 kg of U^{233}.

NUCLEAR PERFORMANCE OF A REFERENCE DESIGN REACTOR

A conceptual design study of a 240-Mw (electrical) central-station molten-salt-fueled reactor (MSR) was described by the Molten-Salt Reactor Group at ORNL.[16] The system employs a two-region homogeneous reactor having a core approximately 8 ft in diameter and a blanket 2 ft thick.

The core, with its volume of 113 ft^3, is capable of generating 600 Mw of heat at a power density in the core of 187 w/cm^3. The general arrangement of the core and blanket is shown in Fig. 7. The basic core salt is a mixture of lithium, beryllium, and thorium fluorides, together with sufficient fluoride of U^{235} or U^{233} to make the system

Fig. 5. Critical Masses of Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}.

[16]J. A. Lane, H. G. MacPherson, and F. Maslan (eds.), *Fluid Fuel Reactors*, p 569, Addison-Wesley, Reading, Mass., 1958.

SECTION A-A

FUEL PUMP
MOTOR

BLANKET
PUMP
MOTOR

BLANKET EXPANSION
TANK

SIPHON DRAIN

FUEL LINE TO
HEAT EXCHANGER

FUEL EXPANSION
TANK

2 0 2 4

FEET

CORE
MOLTEN SALT
FUEL

BREEDING BLANKET

FUEL RETURN

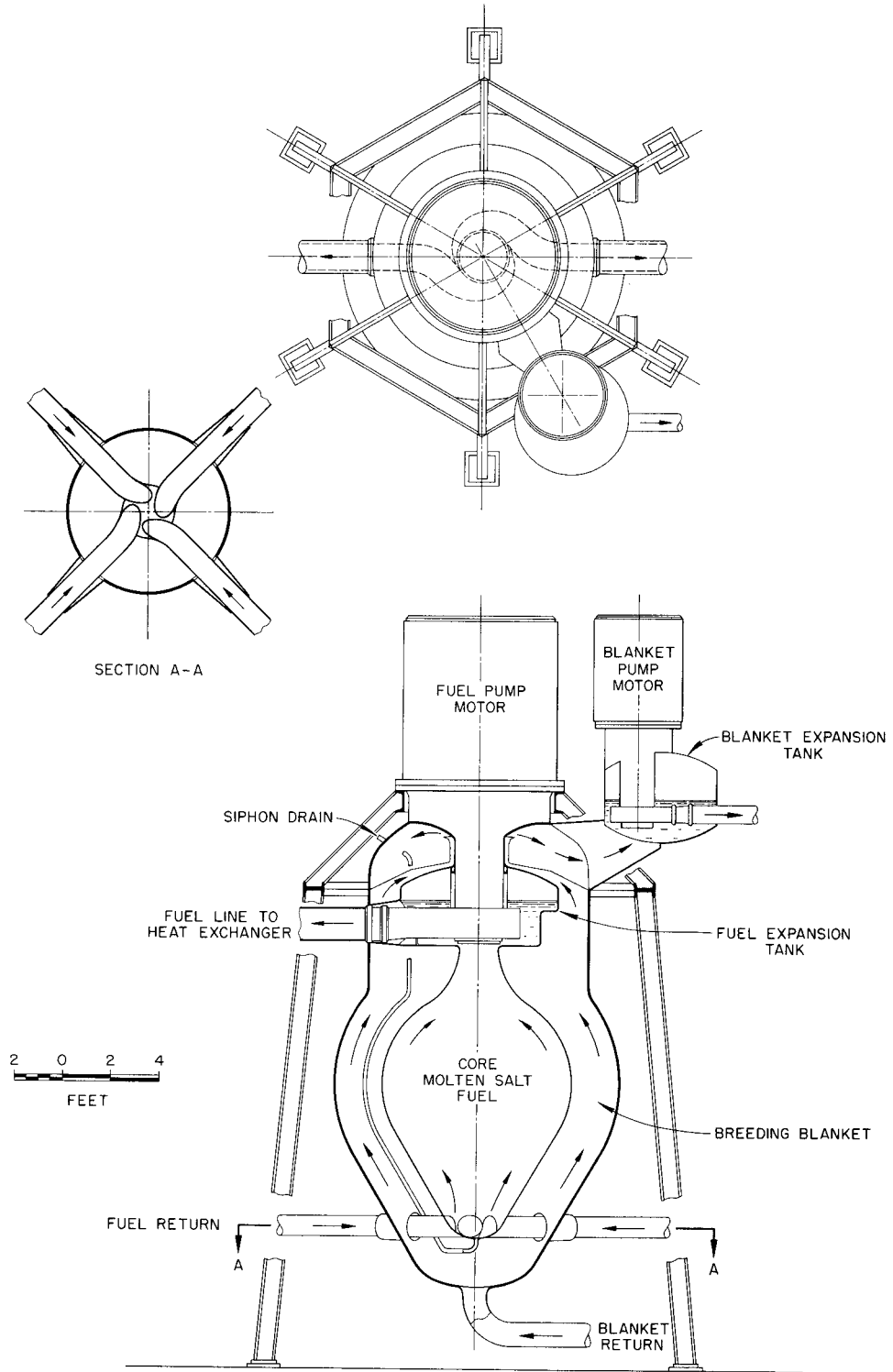A   A

BLANKET
RETURN

Fig. 7. General Arrangement of Core and Blanket.

critical. The blanket contains ThF_4, either as the eutectic of LiF and ThF_4, or mixtures of it with the basic core salt. The liquidus temperature of the fuel salt is about 850°F and that of the blanket is 1080°F or lower.

Both the core fuel and the blanket salt are circulated to external heat exchangers, six in parallel for the core and two in parallel for the blanket. The heat is transferred by intermediate fluids from these heat exchangers to the boilers, superheaters, and reheaters. The heat transfer system is designed so that, with a fuel temperature of 1200°F, a steam temperature of 1000°F at 1800 psi can be achieved.

The volume of fuel salt external to the core in the transfer lines, pumps, and heat exchangers was estimated to be 339 ft^3. It is this external volume that largely determines the fuel inventory of the system. A parametric study of the regeneration ratio as a function of critical inventory in this system was performed. With U^{235} in reactors employing core and blanket salts No. 1, the results are as shown in Fig. 8, where regeneration ratio is plotted vs critical inventory, with thorium concentration in the fuel as a parameter. The numbers associated with the plotted points are the diameters of the cores.

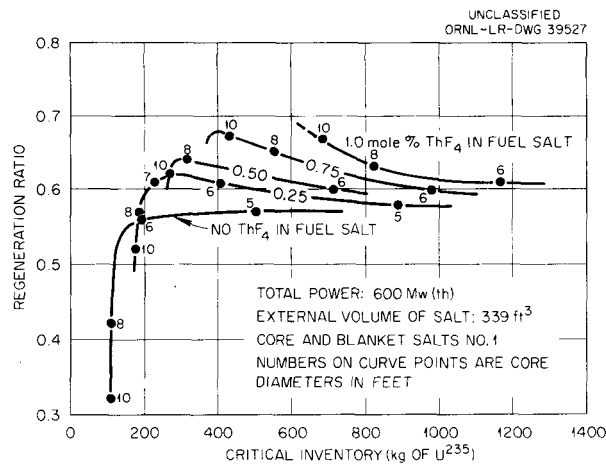The curves are observed to peak rather sharply, and these peaks define a locus of maximum regeneration ratio for a given inventory. It may be seen that, with no thorium in the core, a regeneration ratio of 0.5 can be obtained with an inventory of 100 kg of U^{235} in a 7-ft-dia core. The addition of 0.25 mole % thorium to the core salt yields a regeneration of about 0.6 for an inventory of 200 kg in a 7-ft-dia core. Optimum core size increases hereafter. Also the rate of increase of regeneration ratio falls off substantially. With 0.75 mole % thorium, a regeneration of 0.67 is obtained with 400 kg of fuel in a 10-ft-dia core. As mentioned above, it was felt that it would be difficult to fabricate reliable core vessels having diameters greater than 10 ft, and, accordingly, larger cores were not investigated. However, an examination of the curves indicates that further increases in thorium loading and core diameter would probably not increase the regeneration ratio above 0.7.

Reactors employing core and blanket salts No. 2 (see Table 4) require somewhat lower inventories than the corresponding cores using salts No. 1, as shown in Fig. 9. The 8-ft-dia core, for instance, requires only about 600 kg of U^{235} with a loading of 1 mole % thorium, whereas with the No. 1 salts, about 850 kg was required. The regeneration ratios, however, are about the same, ranging from 0.62 to 0.64 for the 8-ft-dia cores.
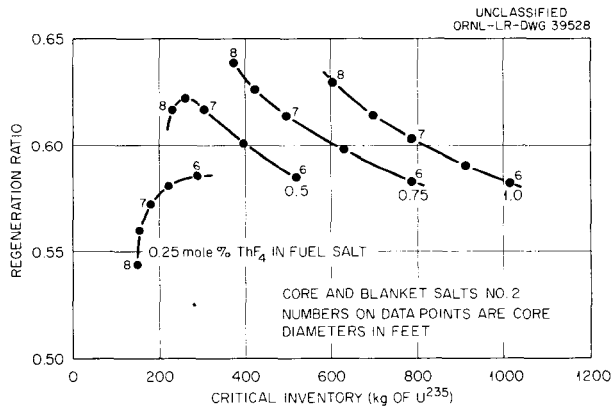

Fig. 9. Initial Fuel Regeneration in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{235}.

With U^{233} as the fuel, there is a marked improvement in the performance, and the inventories are much lower. The performance of cores using core and blanket salts No. 1, with thorium concentrations ranging up to 1 mole %, is shown in Fig. 10. The regeneration ratios range up to 0.95

Fig. 8. Initial Fuel Regeneration in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{235}.

at inventories less than 300 kg of U^{233}. The beryllium-rich core and blanket salts (No. 2) gave substantially the same results, as shown in Fig. 11.
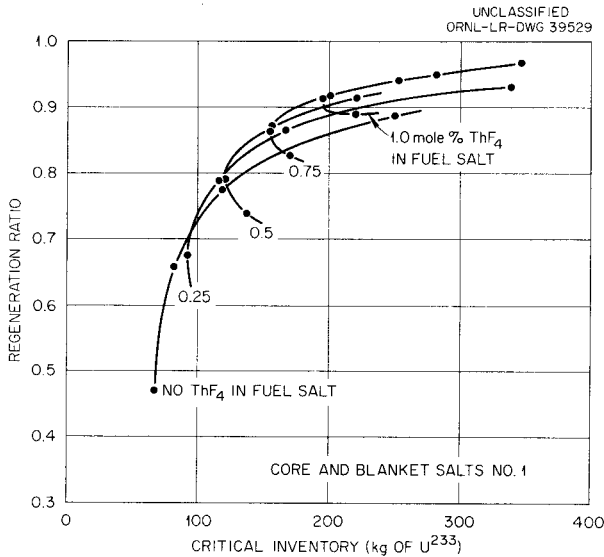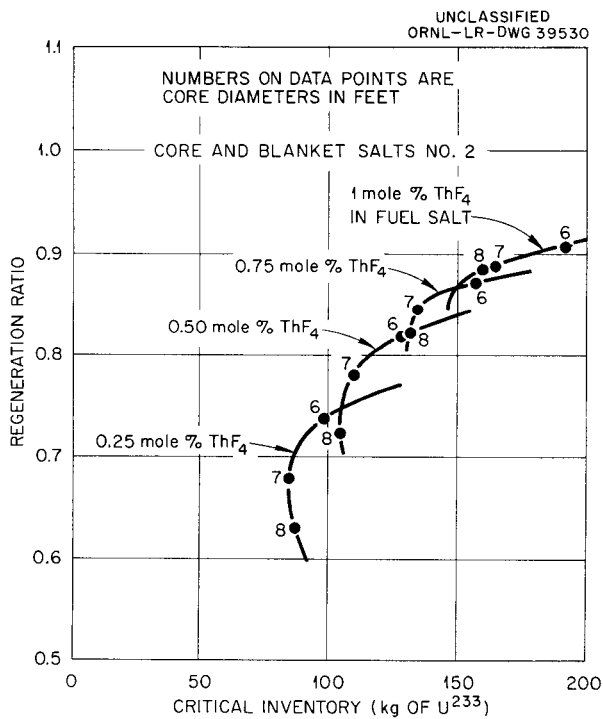
Regeneration ratios of the order of 0.6 can be obtained at inventories of about 100 kg of U^{233}.

Increasing the thorium concentration up to 7 mole % gives a monotonically increasing regeneration ratio, up to about 1.09, but the fuel inventories become very high. The performance of cores having diameters ranging from 4 to 12 ft and thorium concentrations of 1, 2, 4, and 7 mole % are shown in Fig. 12. The dashed line is the estimated envelope of the curves shown and represents the locus of maximum regeneration for a given inventory. It is seen that regeneration ratios above 1.0 can be obtained from fuel investments of 400 kg or greater. Also, it appears that the 8-ft-dia cores give about the highest regeneration at all thorium concentrations. In Fig. 13 the performances of 8-ft-dia cores using fuel and blanket salts No. 2 and U^{235} and U^{233} fuel, respectively, are compared. With U^{235}, a maximum regeneration of about 0.65 is obtained at an inventory of about 400 kg. The same amount of U^{233} gives a regeneration ratio of 1.0, and 1000 kg of U^{233} gives a regeneration of 1.07.

Fig. 10. Initial Regeneration of Fuel in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}.

Fig. 11. Initial Regeneration of Fuel in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}.
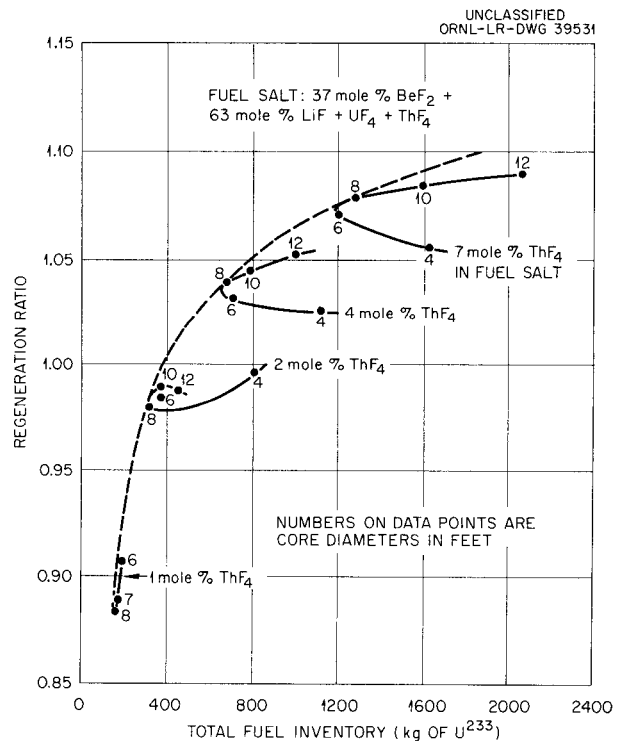

Fig. 12. Initial Regeneration Ratio in Two-Region, Homogeneous, Molten-Fluoride-Salt Reactors Fueled with U^{233}.

17

COMPARATIVE PERFORMANCE

In comparison with converter reactors, the homogeneous MSR fueled with U^{235} is somewhat inferior
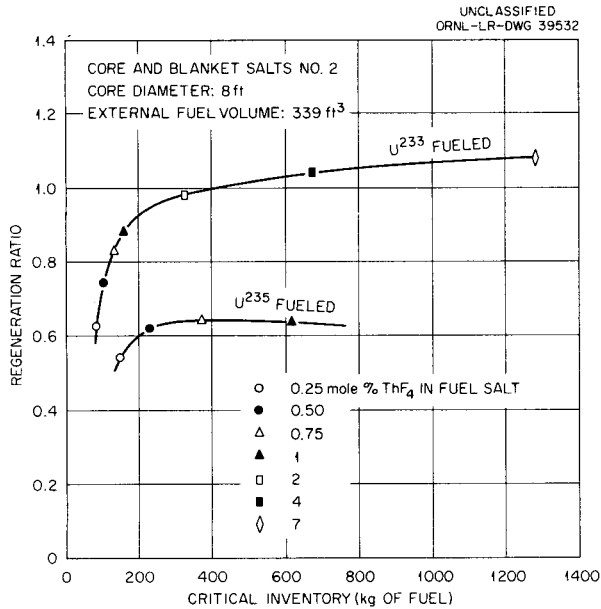

Fig. 13. Initial Regeneration of Fuel in Two-Region, Homogeneous, Molten-Fluoride-Salt, Reference Design Reactors.

in respect to regeneration ratio, but it is capable of matching or exceeding the other systems in specific power, as indicated in Table 5. However, it should be noted that the U^{233} produced in the MSR is recycled, whereas the plutonium produced by the other systems listed is not recycled. The economics of those systems depends strongly on the market value of Pu^{239}, and this isotope is inferior as a fuel relative to U^{233}. Thus a regeneration of 0.5 for U^{233} (at $15 a gram) is equivalent to a regeneration 0.62 for plutonium (at $12 a gram). It is concluded that the molten-salt-fueled system is competitive with solid-fuel converter reactors burning U^{235} in respect to specific power and fuel regeneration. The choice between the two systems therefore lies in other factors, such as reliability, maintainability, and cost of fuel reprocessing.

Table 5. Comparison of Converter Reactors

Reactor*	Power [Mw(e)]	Steam Pressure (psig)	Steam Temperature (°F)	Thermal Efficiency (%)	Fuel Enrichment (% U^{235})	Fuel Inventory (kg of U^{235})	Value of Fuel ($) $\times 10^6$	Specific Power [Mw(e)/million dollars]	Average Conversion Ratio	Isotope Produced
EFR	94	600	740	31.3	28	485	8.0	11.7	1.12	Pu^{239}
GCR	225	950	950	32.1	2	2740	30	7.5	0.74	Pu^{239}
SGR	100	800	825	32.4	3.5	825	10.7	9.3	0.55	Pu^{239}
OMR	150	415	550	26.1	1.5	1280	12.4	12.1	0.73	Pu^{239}
DPR	180	950	540	28.7	1.5	782	7.6	23.6	?	Pu^{239}
ERR	22	600	825	30.2	>90	148	2.5	8.8	0.52	U^{233}
PWR	60	600	486	26.7	>90 0.72	75 96	1.81	33.2	0.7	Pu^{239}
YER	134	520	Saturated	26.2	2.6	550	6.6	20.4	0.64	Pu^{239}
CER	140	600	485	34.7	>90	275	4.7	29.8	0.47	U^{233}
MSR	260	1800	1000	40.5	>90	400	6.8	38.2	0.6–0.8**	U^{233}

*EFR, Enrico Fermi Reactor, *Solid Fuel Reactors*, J. R. Dietrich and W. H. Zinn (eds.), Addison-Wesley, Reading, Mass., 1958.

GCR, ORNL Gas Cooled Reactor, *ibid.*

SGR, Sodium Graphite Reactor, *Sodium Graphite Reactors*, C. Starr and R. W. Dickinson (eds.), Addison-Wesley, 1958.

OMR, Organic Moderated Reactor, *Solid Fuel Reactors, loc. cit.*

DPR, Dresden Nuclear Power Reactor, *Boiling Water Reactors*, A. W. Kramer (ed.), Addison-Wesley, 1958.

ERR, Elk River Reactor, *ibid.*

PWR, Pressurized Water Reactor, *Shippingport Pressurized Water Reactor*, R. T. Bayard *et al.*, Addison-Wesley, 1958.

YER, Yankee Atomic Electric Co. Reactor, *Preliminary Hazards Summary Report*, YAEC-60 (1957).

CER, Consolidated Edison Reactor, *Nuclear Reactor Data No. 2*, Raytheon Mfg. Co., Waltham, Mass., 1956.

MSR, Molten Salt Reactor, *Fluid Fuel Reactors*, J. A. Lane, H. G. MacPherson, and F. Maslan (eds.), Addison-Wesley, 1958.

**Ratio depends on processing rate.

INTERNAL DISTRIBUTION

1-10. L. G. Alexander
11. E. S. Bettis
12. F. F. Blankenship
13. E. P. Blizard
14. A. L. Boch
15. W. F. Boudreau
16. G. E. Boyd
17. M. A. Bredig
18. E. J. Breeding
19. R. B. Briggs
20. W. E. Browning
21. D. O. Campbell
22. W. H. Carr
23. D. A. Carrison
24. G. I. Cathers
25. R. A. Charpie
26. H. C. Claiborne
27. F. L. Culler
28. J. H. DeVan
29. W. K. Ergen
30. J. Y. Estabrook
31. D. E. Ferguson
32. A. P. Fraas
33. W. R. Grimes
34. E. Guth
35. H. W. Hoffman
36. W. H. Jordan
37. P. R. Kasten
38. G. W. Keilholtz
39. C. P. Keim
40. M. T. Kelley
41. F. Kertesz
42. B. W. Kinyon
43. M. E. Lackey
44. J. A. Lane
45. R. N. Lyon
46-47. H. G. MacPherson

48. W. D. Manly
49. E. R. Mann
50. L. A. Mann
51. W. B. McDonald
52. H. J. Metz
53. R. P. Milford
54. J. W. Miller
55. K. Z. Morgan
56. G. J. Nessle
57. A. M. Perry
58. P. M. Reyling
59. J. T. Roberts
60. M. T. Robinson
61. M. W. Rosenthal
62. H. W. Savage
63. A. W. Savolainen
64. A. J. Shor
65. M. J. Skinner
66. J. A. Swartout
67. A. Taboada
68. R. E. Thoma
69. M. Tobias
70. D. B. Trauger
71. F. C. VonderLage
72. G. M. Watson
73. A. M. Weinberg
74. M. E. Whatley
75. G. D. Whitman
76. G. C. Williams
77. C. E. Winters
78. J. Zasler
79-82. ORNL — Y-12 Technical Library,
Document Reference Section
83-102. Laboratory Records Department
103. Laboratory Records, ORNL R.C.
104-105. Central Research Library

EXTERNAL DISTRIBUTION

106. F. C. Moesel, AEC, Washington
107. Division of Research and Development, AEC, ORO
108-695. Given distribution as shown in TID-4500 (14th ed.) under Reactors—Power category
(75 copies — OTS)