LA-3987-MS

LOS ALAMOS SCIENTIFIC LABORATORY
of the
University of California
LOS ALAMOS • NEW MEXICO

Quarterly Status Report on the Space Electric Power R and D Program for the Period Ending July 31, 1968

Part II

(Title Unclassified)

UNITED STATES
ATOMIC ENERGY COMMISSION
CONTRACT W-7405-ENG. 36

AEC RESEARCH AND DEVELOPMENT REPORT

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LA-3881-MS (Part I)
LA-3882-MS (Part II, Conf. RD)
LA-3941-MS (Part I)
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C-93b, ADVANCED CONCEPTS FOR FUTURE APPLICATION--CONVERSION DEVICES
M-3679 (56th Ed.)

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Report distributed August 27, 1968

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THERMIONIC POWER SYSTEMS

ZrH_x-Moderated Thermionic Reactors

General Design Considerations

The scoping study of ZrH_x-moderated thermionic reactors has been essentially completed except for investigations of reactors containing ^{233}U and ^{239}Pu. Included in the study are designs with either fueled (seeded) moderator or with pure ZrH_x (nonseeded) moderator. The thermionic diodes for these designs are assembled in "flashlight" type thermionic fuel rods, and the moderator consists of ZrH_x clad with Hastelloy-N. Dimensions of fuel- and moderator-rod components are given in Table I. In general, the fuel in the thermionic fuel rods was assumed to consist of $^{235}UO_2$-Mo cermet (57 v/o UO_2, 40 v/o Mo, 3 v/o Y_2O_3, 85% dense), but comparative calculations for bulk UO_2 were also made. Tungsten-clad emitters and Nb collectors were used. The fuel in seeded moderator rods was assumed to be ^{235}U-Zr alloy.

An axial cross section of a conceptual design of a ZrH_x-moderated thermionic reactor is shown in Fig. 1. The fuel and the moderator rods are supported by grid plates and by the pressure vessel, both of which consist of Nb. The radial reflector is made of metallic Be and contains poison-backed control drums. Axial reflectors consist of Be and are extensions of the fuel and moderator rods; the triple-layer insulation extends along the full length of the fuel rods. The volume fraction of the coolant (circulating NaK) in the axial reflector regions is the same as in the reactor core.

TABLE I. DIMENSIONS OF THERMIONIC FUEL RODS AND MODERATOR RODS

Thermionic Fuel Rod*	Outside Radius, in.
Fuel	0.265
Emitter	0.270
Cs gap	0.280
Collector	0.310
Insulator	0.330
Sheath	0.340
Moderator Rod (non-seeded)	
ZrH_x	0.346
Gap - SCB seal	0.350
Cladding	0.360
Moderator Rod (seeded)	
ZrH_x	0.335
Gap - SCB seal	0.340
Cladding	0.350

*The axial distance between adjacent emitters in the fuel rods is 0.472 in.

Figure 2 is a radial cross section of a typical thermionic reactor. The fuel and moderator rods are in hexagonal arrays. Void spaces near the edge of the core are filled with partial rods of BeO clad with stainless steel. Reactivity-limited

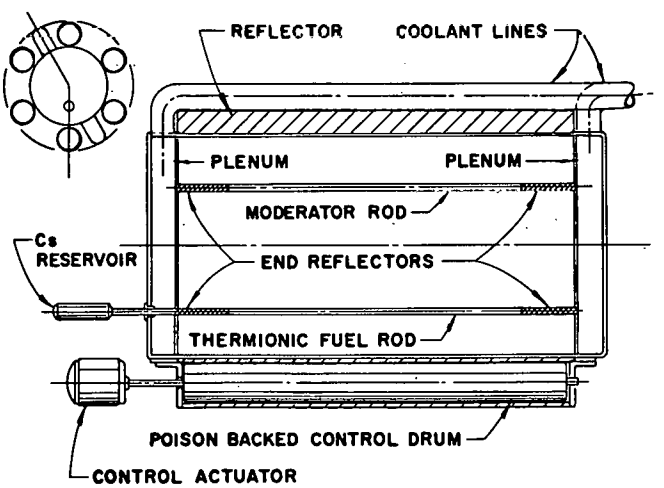

Fig. 1. Axial cross section of ZrH_x-moderated thermionic reactor.

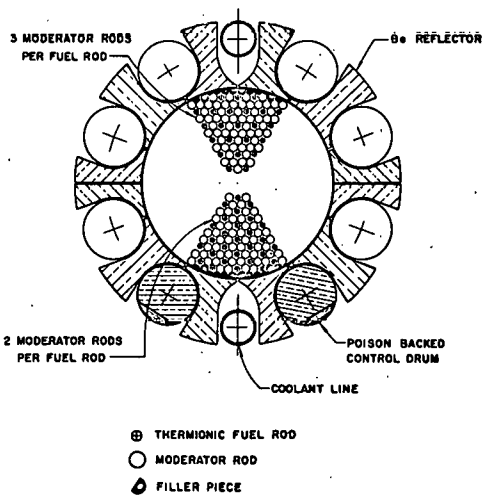

Fig. 2. Radial cross section of ZrH_x-moderated thermionic reactor

reactors of minimum size include from one to three times as many moderator rods as fuel rods. Because these two types of rods should be arranged as symmetrically as possible and because fuel rods should not be placed adjacent to each other, reactor designs were restricted to arrays of either two or three moderator rods per fuel rod. Both of these arrays are shown in Fig. 2; actual moderator-to-fuel rod ratios differ from integral values because of irregularities near the edges of the cores.

Neutronic calculations were made with one-dimensional DTK codes using the S_4 approximation. Initial room-temperature calculations were done with 18 neutron energy groups, four of which were in the thermal neutron energy range, whereas calculations at reactor operating temperatures were done with 16 neutron energy groups. To facilitate calculational procedures, all neutron cross-section sets were collapsed to 16 neutron energy groups for survey calculations.

Because the reactor diameter was assumed to be the most important dimension for mission-oriented power supplies, the radial geometry of each design was determined accurately, and the lengths of the cores or axial reflectors required were calculated for desired values of excess reactivity. These lengths may be slightly in error, but can be adjusted easily without significantly affecting the overall designs. Since an objective of the study was to determine minimum-size reactors, power flattening was not a requirement.

The relevance of the data available for predicting the electric power output of thermionic reactors is, to some degree, in doubt. A method (see LA-3942-MS) was used in this study which arrived at output predictions that are probably conservative and may underestimate current capabilities.

In addition to determining reactor sizes and power levels, weight estimates were obtained for shadow shields and for heat-pipe radiators. The weights of shadow shields were estimated by scaling the weight of the shield for a man-rated SNAP-8 Reactor (see NAA-SR-11984) proportionately to the shield volumes of the thermionic reactor designs. Shield-volume calculations for the SNAP-8 reactor and for each of the thermionic reactors were based on a cone half-angle of 6° and on a gallery height of 9.4 in.; a separation distance of 150 ft was implied. The weights of the heat-pipe radiators were obtained by applying appropriate scaling factors to the weight of the thermionic SNAP radiator (see LA-3867). This radiator design was not optimized for minimum weight; a radiator optimization study is in progress.

Seeded-Moderator Designs

Fuel is added to the moderator of thermionic reactors primarily to reduce the size of reactivity-limited designs. The most obvious disadvantage of this concept is the decreasing overall efficiency as fuel is added to the moderator and, thus, the requirement for proportionately larger radiators and pumps. Furthermore, there are restrictions on power density and coolant volume fraction, as will be discussed later. However, in addition to small size, seeded-moderator designs offer other advantages, such as large negative temperature coefficients, and, due to the fuel in the moderator, another degree of freedom which may be used for partial flattening of the power-density distribution, even in very small reactors.

The SNAP research and development program has provided a large reservoir of proven technology relating to fueled ZrH_x; this was used extensively to establish design limitations for the present study. Two fuel concentrations in the ZrH_x moderator were investigated, namely, 3 and 5 w/o of the Zr (the standard SNAP fuel concentration is 20 w/o of the Zr). The only fuel considered in the moderator was ^{235}U. Limitations on power density and coolant volume fraction were established by comparison with the proposed operating conditions for the 600-kW SNAP reactor containing 241 fuel elements. Specifically, the maximum temperature of the seeded moderator was not allowed to exceed the maximum fuel temperature in the SNAP reactor, and the minimum spacing between adjacent seeded moderator rods was the spacing between SNAP fuel elements scaled linearly according to the relative heat flux from the cladding. The coolant volume fractions associated with the fuel rods and with the moderator rods were 19 and 14 v/o, respectively.

No attempt was made to establish precise excess-reactivity requirements for the designs considered. Upper limits for 10,000 h of operation were estimated for each of the two groups of reactor designs characterized by the concentration of fuel in the moderator. These upper limits are listed in Table II

Table II. Upper Limits on Excess Reactivity Requirements for 10,000 h Operation

	Excess Reactivity Requirement, %	
	3 w/o Fuel in Moderator	5 w/o Fuel in Moderator
Equilibrium Xe	1.44	0.96
Other fission products	0.44	0.33
Burnup	1.20	0.70
Hydrogen leakage	0.40	0.40
Cold to hot operating condition	1.50	2.00
Totals	4.98	4.39

Calculations of excess reactivity requirements due to Xe, other fission products, and burnup were based on the assumption that the reactor power-density distributions are flat. Actually, however, these distributions are not flat, and the calculated values were therefore increased by a factor of 2 to allow for this. Excess reactivity requirements due to H_2 leakage were calculated on the basis of current SNAP technology. There may be a small additional reactivity requirement due to H_2 redistribution caused by thermal gradients in the moderator, but this effect was not evaluated.

Reactor control was assumed to be provided, at least in part, by poison-backed rotating drums. The effect of the control drums on reactivity was not included in the neutronics calculations; the presence of the control poison will reduce the calculated values somewhat (~ 1 to 2%), even with the drums in their most reactive positions.

A total excess reactivity requirement of 6 or 7% appears to be adequate for the seeded designs.

Reactor sizes corresponding to k_{eff} values of 1.05 and 1.10 were determined for 3 and 5 w/o seeded designs. The results of these calculations are presented in Table III.

Nonseeded Moderator Designs

Nonseeded moderator designs are somewhat simpler than those with seeded moderators. Moderator rod sizes and center-to-center spacings of fuel and moderator rods were based on core coolant requirements. Detailed calculations revealed that the power density in nonseeded moderator reactors is trivially small (< 1.5 W/cm^3) and that, therefore, requirements for moderator coolant are minimal. Moderator rod diameters were chosen so that adjacent moderator rods are in contact. The coolant volume fractions associated with fuel and moderator rods were 19 and 9.5 v/o, respectively.

Excess reactivity requirements were not calculated. They are less than for seeded moderator designs, and ~ 5% should be adequate. According to calculations, nonseeded moderator designs have one neutronic characteristic quite different from those with seeded moderators: the changes in reactivity in nonseeded designs are positive when going from room temperature to operating temperature and are due to the effects of thermal neutron spectrum changes. In this temperature range, the Doppler effect and reactivity coefficients due to material expansion are negative, but they are not large enough to compensate for the spectrum coefficient. This does not appear to present a problem since the spectrum coefficient is very sensitive to small changes in design and can apparently be made arbitrarily small. The thermal neutron spectrum calculations used in this survey are not sufficiently detailed to warrant extensive investigation of these effects.

The results of the calculations for nonseeded moderator designs are summarized in Table IV.

Other Fuel Forms

The fuel in the thermionic diodes of the reactors discussed above was Mo-UO_2 cermet.

TABLE III. SEEDED ZrH_x-MODERATED THERMIONIC REACTORS

k_{eff}	No. TFR[a]	No. Mod. Rods	w/o Seed in Mod.	OR,[b] in.	H,[c] in.	Electric Power[d], kW	Overall Efficiency, %	Reactor wt, lb	Radiator wt, lb	Shield[e] wt, lb	Specific wt. with Shield, lb /kWe	Specific wt. w/o Shield, lb /kWe
1.05	73	138	5	10	24.4	33	5.1	746	677	2364	114	43
1.05	73	138	3	10	28.5	42	6.0	828	713	2500	97	37
1.05	73	138	5	9	30.5	46	5.1	888	932	2254	89	40
1.05	55	156	5	10	20.3	19	3.8	618	517	2231	179	60
1.05	55	156	3	10	24.4	25	4.7	739	556	2364	146	52
1.05	55	156	5	9	24.4	25	3.8	627	688	2061	135	53
1.10	73	138	5	10	32.6	50	5.1	1031	1015	2639	94	41
1.10	73	138	3	10	42.8	71	6.0	1359	1213	3007	79	36
1.10	55	156	5	10	24.4	25	3.8	769	688	2362	154	58
1.10	55	156	3	10	30.5	34	4.7	953	766	2568	125	50
1.10	55	156	5	9	28.5	34	3.8	798	946	2188	115	51

[a] TFR - thermionic fuel rod

[b] OR - outside reactor radius including reflector

[c] H - reactor height including reflector but not including coolant plenums

[d] kW - average electric power flux 3.3 W/cm^2, thermionic efficiency 9.5%.

e - SNAP-8 man-rated shield design.

TABLE IV. NON-SEEDED ZrH_x-MODERATED THERMIONIC REACTORS

k_{eff}	No. TFR[a]	No. Mod. Rods	OR,[b] in.	H,[c] in.	Electric Power,[d] kW	Reactor wt., lb	Radiator wt, lb	Shield wt.[e], lb	Specific wt. with Shield, lb /kWe	Specific wt. w/o Shield, lb /kWe
1.05	121	246	11.7	26.4	62	1184	649	2988	78	30
1.05	139	282	12.1	24.4	63	1180	663	3052	79	29
1.05	199	384	13.5	20.3	68	1225	712	3418	104	29
1.05	121	330	12.5	30.5	76	1636	793	3445	78	32
1.05	151	432	13.5	28.5	86	1823	900	3774	76	32
1.10	121	246	11.7	47.6	110	1929	1154	3542	60	28
1.10	139	282	12.1	34.6	103	1766	1077	3457	61	28
1.10	199	384	13.5	24.4	91	1532	949	3582	67	27
1.10	151	432	13.5	40.8	137	2729	1440	4277	62	30

[a] TFR - thermionic fuel rod

[b] OR - outside reactor radius including reflector

[c] H - reactor height including reflector but not including coolant plenums

[d] kW - average electric power flux 3.3 W/cm^2, efficiency 9.5%.

e - SNAP-8 man-rated shield design.

Neutronic calculations were also made, however, for some of these reactors assuming they contained W-clad bulk UO_2 thermionic-diode fuel of 85% theoretical density. All dimensions except the W cladding thickness remained unchanged. The claddings considered were 20- and 40-mil thick natural W and 40-mil thick ^{184}W. The results of these calculations are summarized in Table V.

TABLE V. REACTIVITY CHANGES DUE TO REPLACEMENT OF CERMET THERMIONIC FUEL WITH BULK UO_2

No. TFR	No. Mod. Rods	v/o Seed in Mod.	Electric Power kW	Reactivity Changes, $		
				20-mil W	40-mil W	40-mil ^{184}W
73	138	5	50	0.15	- 5.03	3.44
73	138	3	71	0.06	- 6.49	3.72
199	384	0	91	-1.22	-14.39	3.66

General Conclusions

Electric power levels from 60 to 140 kW can be obtained from reactivity-limited nonseeded moderator designs. For a given value of excess reactivity, the specific weights of both shielded and unshielded designs are essentially independent of power level and height-to-diameter ratio. This is due partly to the value of cone half-angle used in obtaining shield weight estimates. Variations in shield weight would be wider if a larger cone half-angle had been used. The insensitivity of the specific weights to design variations is also attributable to the fact that electric power outputs were estimated on the basis of constant power per unit area of emitter and were not affected by differences in power distribution between different reactor designs.

Electric power levels of less than ~ 60 kW can be obtained from nonseeded designs only by derating the performance of these designs. In this case, system weights would probably not change significantly although changes in total thermal power will affect radiator weights. Higher power levels could be obtained from designs similar to those investigated, and specific weights for such designs would decrease as the electric power level is increased.

Seeded-moderator reactors can be designed for electric power levels as low as 20 kW without derating thermionic-rod performance. Such reactors would be superior to designs resulting from SNAP reactors adapted to thermionic conversion, which were studied previously (LA-3867).

At electric power levels of ~ 50 kW the specific weights of seeded and nonseeded moderator designs are about the same, whether the reactors are shielded or not. Below 50 kW, however, the seeded-moderator designs offer some advantage in specific weight. If waste-heat radiators with significantly smaller weights per unit of heat radiated can be designed, the differences in specific weight between seeded and nonseeded moderator designs at low power levels will become more pronounced.

Thermionic Fuel Studies

Fuel Pin Development

Several batches of UO_2 particles were coated with Mo using paraffin as a binder. One series of 19μ average particle size UO_2 powder was coated using varying amounts of paraffin and 2 and 4μ Mo powder. Approximately 30 v/o Mo was retained on the UO_2; microscopic evaluation revealed the Mo coating to be continuous.

Both hydrostatically pressed and sintered and hot-pressed pins were made with 40 and 60 v/o UO_2 loadings. Large variations in the final open porosity of some of the pressed and sintered pins were noted. It was suspected that this could be traced to variations in dewaxing or sintering conditions, but examination of furnace run sheets did not show any differences which would be responsible.

The electrical resistivity of 34 Mo-UO_2 fuel pins was determined. The data showed a linear relationship between electrical resistivity and open porosity for the 40 v/o UO_2 pins. At ~ 6% open porosity, however, the slope of the curve increased drastically (Fig. 3). This change is attributed to the thinning of the Mo coating due

EFFECT OF FUEL LOADING AND % OPEN POROSITY ON THE ROOM TEMPERATURE ELECTRICAL RESISTIVITY OF COLD PRESSED & SINTERED Mo UO_2 CERMETS.
Mo WAS NOMINAL 4.0 μ IN FISHER APS
UO_2 WAS NOMINAL 4.5 μ IN FISHER APS

ALL SPECIMENS WERE BLENDED, HYDROSTATICALY PRESSED AT 20,000 PSI AND SINTERED AS FOLLOWS. 3h - 1700°C H_2 AND 1h - 2200°C - Ar.

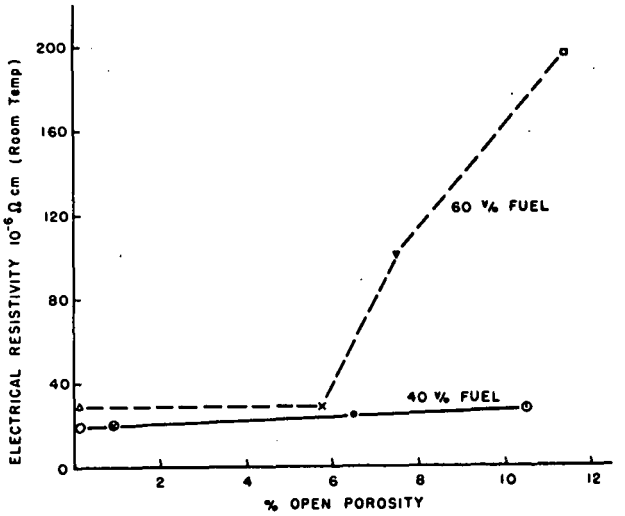

Fig. 3. Effect of porosity of fuel pins on electrical resistivity.

to the combined high UO_2 loading and higher open porosity. This would tend to increase the resistivity of the Mo.

The Mo-UO_2 fuel pins must be coated with W to minimize UO_2 migration and to prevent the fission gases from escaping into the surrounding systems. Some problems have been encountered with lack of adherence of the W coating. This was caused by the high percentage of UO_2 exposed to the surface being coated. One possible solution to the problem would be depleting the surface of UO_2 prior to coating. Pins were heated to 1650-1800°C in vacuum in a cold-wall induction furnace to accomplish this prior to coating. Eight to 12 mils of W were then deposited by the reduction of WF_6 by H_2 at 500°C. After coating, the pins were heated to 2000°C in vacuum to obtain a metallurgical bond between the coating and substrate.

Metallographic examination revealed little difference between the pins depleted at the different temperatures. It does appear, however, that UO_2-depletion temperatures of 1650-1750°C (optical) are adequate to promote bonding between the W coating and pin.

Irradiation Testing of Fuel Pins

Irradiation testing of Capsule LA-26-10, the last of four fuel capsules involved in the small-diameter Mo-UO_2 fuel pin irradiation program, has been completed after a total accumulated irradiation time of 10,400 h. The capsule will be shipped from the MTR to LASL in a few weeks, after its presently high radioactivity has decreased sufficiently.

The accumulation of posttest data on the other three fuel capsules is continuing. A summary of the data obtained so far on fuel porosity, fission-gas escape, dimensional increase, and burnup is given in Table VI. The significance of the dimensional increase was discussed in LA-3882-MS. Posttest measurements of open porosity obtained on fuel specimens from Capsules LA-26-12 and -13 after 6113 and 3245 h of irradiation, respectively, revealed surprisingly little change from pretest measurements. This suggests that the observed volume increases are associated with increases in closed porosity of the cermet, which may explain why less fission gas escaped than expected. This finding is not consistent, however, with the results obtained in earlier irradiation tests (see LA-3462-MS). The difference between the results may be attributable to a more rapid fuel sintering rate arising from the higher average fuel temperatures in the current irradiation-test series.

Burnup data have been obtained for only two of the test specimens. In earlier tests (see LA-3462-MS), percent burnup had been determined by measuring the amount of ^{144}Ce formed during irradiation. These measurements were not only relatively arduous but also somewhat suspect because some of the ^{144}Ce might have been lost due to diffusion during testing. The data presented in Table VI under the heading "fissions/cm^3" were obtained by determining, by a mass spectrometer, the abundance ratio of the ^{235}U, ^{236}U, and ^{238}U isotopes before and after irradiation. Measuring the increase of ^{236}U and the decrease of ^{235}U yielded two independent determinations of

TABLE VI. IRRADIATION-TEST DATA FOR Mo-40 v/o UO_2

CERMET FUEL PINS (10% ENRICHED UO_2)

Test No.	Irradiation Time, h	Pin No.	Particle Size, μ UO_2	Particle Size, μ Mo	Density, % Theoretical	Open Porosity Initial	Open Porosity Post-test	Gas Escape, % UO_2	Gas Escape, % Cermet	Diametral Increase, % Max.	Diametral Increase, % Ave.	Length Increase, %	Burnup, Fissions/cm³	Mean Power Density, W/cm³	Temp., °C
		5128-R_1-10	125	4	87	10	6.3	60	49	.4	.2	0	8.45×10^{19}	301	1345*
LA26-11	2297	5128-R_1-12	125	4	87	10	6.9	55	40	1.0	.6	0.5	8.39×10^{19}	299	1345*
		5117-R_2-15	4	4	91	7	12.7	70	54	5.2	3.6	6.5			(1750)**
		5116-R_2-2	4	4	89	11	10.9	74	38	1.6	.8	-0.7			
LA26-13	3245	5127-R_2-1	125	4	91	7	8.2	59	46	4.8	2.4	1.9	(1.2×10^{20})**	(300)**	(1750)
		5127-R_2-4	125	4	91	7		60	50	4.2	2.2	2.5			
		5127-R_2-6	125	4	91	7	6.8	59	43	4.0	2.4	1.0			
		5116-R_2-17	4	4	89	11	---	--	--	2.4	1.8	1.9			
LA26-12	6113	5116-R_2-14	4	4	89	11	9.6	--	--	3.1	2.0	1.7	(2.2×10^{20})	(300)	(1750)
		5128-R_1-8	125	4	87	10	---	--	--	4.1	2.6	2.4			
		5128-R_1-9	125	4	87	10	9.3	--	--	4.0	2.4	1.7			
		5116-R_2-7	4	4	89	11	---	--	--	---	---	---			
LA26-10	10,400	5116-R_2-14	4	4	89	11	---	--	--	---	---	---	(3.8×10^{20})	(300)	(1750)
		5128-R_1-13	125	4	87	10	---	--	--	---	---	---			
		5128-R_1-11	125	4	87	10	---	--	--	---	---	---			

* Low temperature value due to the presence of H_2 in the fuel test chamber.
** Values of fissions/cm³, mean power density, and temperature which appear in parentheses are based on MTR neutron flux levels and have not yet been verified by burn up measurements.

burnup. The latter method is feasible because the fuel specimens are only ~ 10% enriched and because fission densities on the order of 10^{20} fissions/cm³ result in a ^{235}U burnup of 14%. The probable error at this burnup level is about ± 15%. It will be considerably less for the longer irradiation tests. The burnup determination based on an increasing ^{236}U abundance is accurate to within ± 4%.

Fission-Gas Collection

The mockup of the hot-cell fission-gas collection system described in LA-3942-MS was used in further tests to determine whether ^{85}Kr is held back by any portion of the system, including the dissolving fluid, the flask walls, the Tygon tubing, and the Cu tubing. No evidence of any holdup was detected. The earlier Linde 13X molecular sieve has been replaced by a sieve consisting of equal parts of Linde 4A and 5A material. The efficiency of a single collection can was thus increased from 98 to essentially 100%.

Although the use of molecular sieve has markedly increased the collection efficiency, it also caused some difficulties. The distribution of ^{85}Kr in the collection can is apt to be nonuniform, and γ-absorption by the molecular-sieve material must be taken into consideration, thus complicating the counting procedure. Moreover, ^{85}Kr is being retained by the molecular sieve even at elevated temperature so that a completely independent determination by mass spectrometric measurement of the amounts of Kr and Xe collected is no longer possible.

Development of 0.5-in. Diam Cermet Fueled Emitter

An emitter design which should reduce or eliminate undesirable swelling of cermet fuel pins was described in detail in LA-3947-MS. In this design cylindrical cermet shapes are slotted so as to form four pie-shaped quadrants which are supported at each end by pressure-bonded Mo tacks and, on the outside, by W-cladding. In principle, each segment has room enough to expand without increasing the o.d. of the emitter. Several Mo-60 v/o UO_2 pins of this type have been made, complete except for W cladding, and are awaiting developments in the W cladding process. While being prepared for cladding (i.e., during depletion of UO_2 from the surfaces) the pins were thermally cycled successfully once to ~1800°C.

EBR-II and OWR Isothermal Irradiator Tests

Irradiation of Bushings

The EBR-II isothermal irradiator experiment is designed to determine the long-term irradiation on small cylindrical bushings made of various insulating materials, namely, Al-995, AD-999, Lucalox (Al_2O_3), and Y_2O_3. The bushings are to be irradiated for 5000 h at 1.3×10^{14} nvt ($E_n > 1$ MeV) and at a nominal temperature of 750°C.

Difficulties have been encountered in the construction of the dewar unit which is used to reduce the radial heat flow from the insulator test chamber. During final preassembly bakeout, the polished Au layer of the inner dewar wall blistered. The Au had been electroplated in several successive layers onto the ceramic Al_2O_3 layer, which prevents the Au from diffusing into the stainless-steel base. The blistering was due to volatilization of contaminants entrapped between the various Au layers.

To avoid the blistering and other fabrication problems, the inner dewar wall is now made of Ni. If highly polished, the Ni-Au dewar has thermal insulating properties almost equal to those of the Au-Au combination and is much easier to fabricate. The chief disadvantage of the Ni-Au system is that its insulating capability is almost entirely dependent on the high reflectivity of the Au layer on the outer dewar wall.

Because of large discrepancies between calculated and measured γ-heating rates in the EBR-II, there is some doubt as to whether enough heat will be generated in one of the isothermal irradiator assemblies for proper temperature regulation. To provide a margin of safety in the amount of heat generated, the stainless-steel holding caps used to retain the insulator specimens on the heat-pipe assembly have been replaced with more massive caps of Mo and W. There will then be an additional 50 W of γ heating; this represents a 50% increase and should be adequate to compensate for any uncertainties in design.

The EBR-II experiment is now completely assembled and checked out except for the end caps on either end of the enclosing thimble. These caps are being welded in place.

Two mockups of the EBR-II experiment, OWR-I and OWR-II, are still being irradiated in the Omega West Reactor. Accumulated irradiation times for the two mockups are 5305 and 2266 h, respectively.

Tri-Layer Assembly Development

Autoclaving of Tri-Layer Assemblies

Nine Nb-1% Zr/Al_2O_3 (Lucalox)/Nb-1% Zr tri-layer assemblies have been autoclaved to date in three separate runs. The assemblies did not bond adequately, and the sheath insulators cracked severely. Three of these assemblies were autoclaved at 1460°C and 10,000 psi for 3 h and were then submitted to a 10-h cooling procedure to minimize the stresses induced in the sheath insulators by the different coefficients of thermal expansion of the Nb and the Al_2O_3 (see LA-3942-MS). Only one of these three samples did not crack. Metallographic examination revealed that a good mechanical bond had been obtained, but that little, if any, of the desired diffusion-bonding had occurred. Other recent investigations have shown that part, but by no means all, of the cracking problem has been caused by a lack of sufficiently gentle handling during assembly.

Four new tri-layer assemblies were prepared for autoclaving. These assemblies reflected the unit modular design in thermionic configurations, in which the insulator is 2.5-in. long. Three of the assemblies had Lucalox insulators with wall thicknesses of 0.020 in. (two units) and 0.030 in. (one unit), and one assembly had an AD-999 (Coors) insulator with a wall thickness of 0.030 in. Another, longer assembly, designed for use in a tri-layer irradiation test, was prepared with a 5½-in. long Lucalox insulator having a 0.020-in. thick wall.

All five assemblies were autoclaved by an outside vendor at 1650°C, instead of 1460°C, in an attempt to induce diffusion bonding by increasing the temperature. The pressure remained unchanged (10,000 psi). These assemblies are being evaluated.

Diffusion Bonding of Tri-Layer Assemblies

The possibility is being investigated of stacking tri-layer assemblies by diffusion bonding. Several materials (Ni, Mo, and Zr) are considered usable for promoting the bonding of Nb-Nb interfaces. Pressure, temperature, and diffusion rate of thin layers of the bond-promoting materials, together with surface preparation of the mating Nb-1% Zr cylinders, are the parameters being studied. Trial bonds are evaluated by testing for vacuum tightness with a He leak detector and by metallographic and electron-microprobe examination.

Initially, Nb-1% Zr cylinders with 0.065-in. thick walls, were polished thoroughly to obtain a smooth mating surface after machining. It has since been found that the test pieces have as-machined surface finishes of 3-8 μin., as determined by a profilometer gauge, so that polishing is unnecessary. The test pieces normally receive an outgas firing at 1150°C for 10 min which produces a hardness of 80 ± 5 DPH (diamond pyramid hardness).

Bonding is accomplished by placing the test pieces in vacuum, applying pressures of 5000 to 10,000 psi by means of an externally mounted hydraulic system, and then bringing the temperature up to the desired value. A die set is used to maintain sample alignment and to aid in uniform load application.

Heating is accomplished by an rf induction coil closely coupled to the work piece; temperatures are recorded by thermocouples and by optical means. These temperatures are low--800, 900, and 1000°C--to remain compatible with the eventual stacking of tri-layer subassemblies. The diffusion-promoting heat soaks after bonding have also been kept in this range.

To date, Ni foil (0.0001-in. thick washers) and electroplated Ni (0.0003-in. thick on mating faces) have shown the greatest reliability for initial bonding at the low temperatures and pressures. The Ni foil, when heated before applying pressure, pulled itself from the joint area. A pressure of 5000 psi applied prior to and during heating the specimens to 900°C for 30 min resulted in leak-tight bonds. These joints were He-tight after 5 h additional heating (without pressure) at 900°C. The samples are currently undergoing metallographic examination.

Bonding has also been obtained with 0.003-in. thick Zr and Mo foils at 10,000 psi and 1000°C. However, long-term heat soaks (150 h at 1100°C) resulted in void formation in the region of the joint. The most recent test using as-machined Nb-1% Zr with 0.0003-in. thick Zr at 900°C and 5000 psi left a porous joint.

HEAT PIPE SYSTEMS

The results obtained in the work on Heat Pipe Systems are unclassified and are described in Space Electric Power R and D Program Report, Part I, LA-3986-MS.

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