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Quarterly Status Report on the
Space Electric Power R and D Program (U)
for the Period Ending January 31, 1972

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QUARTERLY STATUS REPORT ON THE
SPACE ELECTRIC POWER R AND D PROGRAM (U)
For the Period Ending January 31, 1972

SUMMARY

A series of ceramic-metal multilayers is being fabricated for inclusion in EBR-II Isothermal Irradiator Capsules K-5 and K-6. The production of various $Y_2O_3/Nb-1\%Zr$ pentalayer configurations is being emphasized, but specimens using high-grade Al_2O_3 , plasma-sprayed Al_2O_3 , and $Nb-1\%Zr/Al_2O_3$ cermet transition layers are also being produced.

Confirmation has been obtained that crack-free trilayers can be produced by the hot-gas isostatic pressure technique at $1550^\circ C$ even when the alumina layer is put into tension and its diameter is increased by more than 1%. This capability expands considerably the design options for ceramic-metal multilayer fabrication. The use of Al_2O_3 transition layers at $Y_2O_3/Nb-1\%Zr$ interfaces increased the strength of a $Nb-1\%Zr/Y_2O_3$ trilayer under tension; the effect does not appear to be associated with the ceramic-metal bond strength.

Oak Ridge National Laboratory (ORNL) Irradiation Capsule EB-2 containing Al_2O_3 specimens has been disassembled. Gross specimen damage was found to correlate best with impurity content, but dimensional increase and posttest specimen strength correlate with grain size.

Fission-gas retention and ^{235}U burnup measurements were made for the UO_2 -fueled emitters from the General Electric (GE) thermionic converter Test Units SCIP-518, TFE-619, and TFE-621. Retention percentages for ^{85}Kr ranged from 7 to 10% for the first two units listed. Retention measurements for ^{137}Cs also fell within this range. The emitters from TFE-621 operated at considerably higher temperatures, and gas retention was only 2% for the two of the three emitters. The retention value for the third emitter was 24%--a result sufficiently inexplicable to occasion a complete repeat of the measurements.

Uranium-dioxide particles ranging in size from 10 to 25 μm have been coated successfully with tungsten layers less than 1 μm thick. This is the first step in an attempt to produce structurally weak W- UO_2 cermets with very high UO_2 content capable of very high fractional release of fission gases.

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SHEATHED INSULATOR DEVELOPMENT

Fabrication of Multilayers

Multilayer fabrication work has been carried on in parallel programs of producing prototype-size specimens of various configurations for testing in the Experimental Breeder Reactor (EBR-II) and producing specimens for subsequent destructive testing in a series of procedures designed to characterize the nature of the bonding, changes in material properties occasioned by the hot-gas isostatic bonding process, and the relative merits of various assembly technique and materials selections. A series of multilayers is currently being produced for irradiation in the EBR-II Isothermal Irradiator Capsules K-5 and K-6. Compositions and fabrication details of these units are shown in Table I and Fig. 1. The pressure-bonding condition will be 10,000 psi at 1550°C. The specimens include trilayers and pentallayers, and the ceramic choices are yttria and high-grade alumina. The bulk of the effort is concentrated on developing various methods of producing yttria pentallayers.

Stress analysis of the effect of irradiation-induced swelling of alumina on trilayer configurations has shown that the bond between the inner metal

Fig. 1. Longitudinal cross section of cylindrical insulator, showing metal interlayer connected to the collector. The metal interlayer is not present in the large-diameter trilayers. (All dimensions in inches.)

sheath and the ceramic layer is placed in tension that increases linearly with alumina volume change. For this reason, a 50:50 v/o $\text{Al}_2\text{O}_3:(\text{Nb}-1\% \text{Zr})$ layer has been introduced at this interface in some samples, partly to increase the bonding strength but mostly to reduce stress concentrations by expanding the transition region between metal and ceramic. In principle a continuous grading of metal to ceramic would be ideal, but the production of such a layer is a slow and expensive process particularly in pentallayer construction. The purpose of testing the single transition layer is to determine by how much such a relatively simply constructed layer would improve the resistance of the ceramic-metal bond to neutron-damage effects. Trilayer components incorporating closer approximations to a continuously graded transition region are being produced by Gulf General Atomic Co. (GGA) for inclusion in Test Units K-5 and K-6.

Four of the units listed in Table I (LDPLs-133, -137, -138, and -139) have been completed and are being evaluated prior to irradiation. The metal components of the remaining multilayers have been machined, and application of the insulating layers is in progress.

Additional multilayers completed during this reporting period are listed in Table II.

Characterization of Multilayers

Trilayer LDTL-132, described in Table II, was designed to verify an observation that the ceramic layer (Al 995) in LDTL-120 was crack-free even though it had been in tension during pressure-bonding. The inner diameter of the outer sheath of LDTL-132 was made 0.010 in. larger than the outer diameter of the ceramic cylinder, and the wall thickness of the outer sheath was made 3.2 times greater than the collector wall to ensure that the ceramic would be in tension during pressure-bonding. Measurements after pressure-bonding revealed that the inner diameter of the collector had increased by 0.013 in. When a 0.375-in.-wide band of the ceramic was exposed by etching-away the metal envelope, the outer diameter of the ceramic was found to be 0.013 in. larger than before pressure-bonding. No cracks were detected, and the exposed ceramic was leak-tight, as indicated by

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TABLE I
SHEATHED INSULATORS FOR EBR-II
ISOTHERMAL IRRADIATION CAPSULES K-5 AND K-6

Specimen No.	Composition and Thickness, in. ^a				Outer Sheath
	Collector	Insulator			
LDPL-133	Nb-1%Zr 0.067	Y ₂ O ₃ / Nb-1%Zr / Y ₂ O ₃ 0.0075 0.005 0.0075			Nb-1%Zr 0.030
LDPL-134		Y ₂ O ₃ / Nb-1%Zr / Y ₂ O ₃ 0.0075 0.005 0.0075			
LDPL-135		Y ₂ O ₃ / Nb-1%Zr / Y ₂ O ₃ / Al ₂ O ₃ 0.0075 0.003 0.006 0.004			
LDPL-136		(50 v/o Nb-1%Zr / 50 v/o Y ₂ O ₃) / Y ₂ O ₃ / Nb-1%Zr / (50 v/o Nb-1%Zr / 50 v/o Y ₂ O ₃) / Y ₂ O ₃ 0.002 0.0055 0.005 0.002 0.0055			
LDPL-137		Al ₂ O ₃ / Nb-1%Zr / Al ₂ O ₃ 0.0075 0.005 0.0075			
LDPL-138		Al ₂ O ₃ / Nb-1%Zr / Al ₂ O ₃ 0.0075 0.005 0.0075			
LDPL-139		(50 v/o Nb-1%Zr / 50 v/o Al ₂ O ₃) / Al ₂ O ₃ / Nb-1%Zr / (50 v/o Nb-1%Zr / 50 v/o Al ₂ O ₃) / Al ₂ O ₃ 0.002 0.0055 0.005 0.002 0.005			
LDTL-140		(50 v/o Nb-1%Zr / 50 v/o Y ₂ O ₃) / Y ₂ O ₃ / (50 v/o Nb-1%Zr / 50 v/o Y ₂ O ₃) 0.002 0.016 0.002			
LDTL-141		Y ₂ O ₃ 0.020			
LDTL-142		(50 v/o Nb-1%Zr / 50 v/o Al ₂ O ₃) / AD-999X 0.002 0.020			
LDTL-143		AD-999X 0.020			

^aAll ceramic layers are applied by arc-plasma spraying, except the AD-999X in LDTLs 142 and 143. Metal interlayers are sprayed except in LDPLs 134, 138, and 139, which were machined from Nb-1%Zr tubing.

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TABLE II
SHEATHED INSULATORS PRESSURE-BONDED AT 10,000 PSI AND 1550°C

Specimen	Collector	Composition and Thickness, in. ^a				OD, in.
		Insulator		Outer Sheath		
SDPL-125	Nb-1%Zr	Al ₂ O ₃	Nb-1%Zr	Al ₂ O ₃	Nb-1%Zr	0.626
	0.035	0.010	0.003	0.010	0.012	
SDPL-126	Nb-1%Zr	Y ₂ O ₃	Nb-1%Zr	Y ₂ O ₃	Nb-1%Zr	0.526
	0.035	0.010	0.005	0.010	0.012	
SDTL-129 -130 -131	Nb-1%Zr		Y ₂ O ₃		Nb-1%Zr	0.626
	0.035		0.020		0.012	
LDTL-132	Nb-1%Zr		Al-995		Nb-1%Zr	1.336
	0.022		0.020		0.070	

^aThe ceramic and metal interlayers in the small-diameter multilayers were applied by arc-plasma spraying.

a helium-leak test. An ultrasonic scan of the trilayer indicated complete bonding with no voids. These findings indicate that Al₂O₃ can be put in tension during pressure-bonding and that permanent expansion of a cylinder can be accomplished without impairing the vacuum integrity of the ceramic.

Sheathed Insulators SDTLs-128 and -129 were designed to determine the effects of Al₂O₃ as a bond between Nb-1%Zr and Y₂O₃. Insulator SDTL-129, listed in Table II, and Insulator SDTL-128, listed in Quarterly Report LA-4844-MS, p. 8, were similar except for the 0.002-in.-thick layers of Al₂O₃ between the Y₂O₃ and metal sheaths in SDTL-128, which were omitted in SDTL-129. These specimens were cut into 0.5-in.-long cylinders that were sliced longitudinally to form arc segments with chord lengths of 0.375 in. These segments were glued to pull-test adapters with epoxy cement, and, after curing the epoxy, were pulled apart at a pull rate of 0.010 in./min at room temperature. Test data are shown in Table III.

The data, despite their large spread, clearly show that SDTL-128 was stronger than SDTL-129, but the reason for its greater strength has not been determined. The primary mode of failure in each trilayer was fracture in the body of the ceramic and not at the ceramic/metal interface, indicating that the ceramic/metal bond was stronger than the ceramic. This result is still under study.

TABLE III
BOND TENSILE STRENGTH OF YTIRIA TRILAYERS

SDTL-128		SDTL-129	
Specimen No. ^a	Load at Failure, psi	Specimen No.	Load at Failure, psi
1-A	> 2633	1-A	145
2-A	> 2497	2-A	478
3-B	1845	1-B	845
4-B	1580	1-D	1063
5-D	2309	-	-
6-D	2923	-	-
Average	> 2297	Average	633

^aTwo specimens each (e.g., 1-A and 2-A) were cut from each 0.5-in.-long cylinder. Missing segments failed during machining or during subsequent handling.

Despite considerable densification of plasma-sprayed layers during the hot-gas isostatic pressing of pentallayers, results to date in producing crack-free units have been very favorable. This has spurred exploration into other methods of applying the various layers. Of particular interest is the electrophoretic-deposition process where the densities of the deposited layers would be lower than those produced by plasma spraying, but where ease and versatility of fabrication may offer significant advantages. An experimental project to study the production of multilayers by electrophoretic deposition has been initiated.

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NEUTRON-INDUCED DAMAGE TO CERAMICS
AND CERAMIC-METAL COMPONENTS

Disassembly of ORNL Irradiation Capsule EB-2

The Los Alamos Scientific Laboratory has been assigned the responsibility for postirradiation examination and analysis of Oak Ridge National Laboratory (ORNL) Irradiation Capsule EB-2. The capsule was received at Los Alamos on December 10, 1971.

The capsule contained alumina insulator specimens in five subcapsules. Each subcapsule contained six specimens consisting of one cylindrical shell and of one solid cylinder from each of Coors, WESGO, and Opaque Lucalox alumina. All specimens are nominally 0.4 in. in diameter, and the cylindrical shells have 0.020-in.-thick walls. The capsule was irradiated in Position 7D5 of the EBR-II reactor for 7572 Mwd. The design test temperature was 800°C.

The results of spectrographic analyses performed by ORNL of samples of the materials included in the capsule have been received and are summarized in Table IV.

X-ray photographs were taken of the capsule prior to disassembly to aid in establishing the cutting sequence. Disassembly of all subcapsules has now been completed; each insulator specimen was easily removed from the respective subcapsule.

Three stainless-steel neutron-flux monitors and three SiC temperature monitors were included in the capsule. Preliminary estimates of neutron fluences to which the capsule was exposed have been made.

These estimates were based on the number of $^{54}\text{Fe}(n,p)$ ^{54}Mn reactions which occurred in the three stainless-steel flux monitors included in the capsule and indicated that the capsule had been exposed to between 3.2 and 5.8×10^{21} nvt ($E_n > 0.1$ MeV). The axial center of the capsule was thought to have been at the axial midplane of the reactor; however, there is considerable asymmetry in the results obtained from the flux monitors. Because of this, samples of the stainless-steel outer containment are being prepared for analysis to check the estimates given above. The temperature monitors have been shipped to ORNL for interpretation.

Dimensional measurements have been completed for all specimens. The ranges of volume swelling determined by comparison with preirradiation dimensions are given for the various types and shapes of specimens in Table V. These data will be presented in terms of volume increase vs fluence when the additional fluence determinations are completed.

TABLE V

VOLUME SWELLING OF ALUMINA SPECIMENS
FROM ORNL CAPSULE EB-2

Type of Alumina	Shape	Swelling, %
Coors AD-995	Cylindrical shell	3.2 - 7.0
	Solid cylinder	4.7 - 7.5
WESGO Al-995	Cylindrical shell	7.5 - 8.9
	Solid cylinder	8.3 - 10.5
Lucalox	Cylindrical shell	2.8 - 3.6
	Solid cylinder	3.2 - 4.3

TABLE IV

PREIRRADIATION CHARACTERISTICS OF ALUMINA SPECIMENS
INCLUDED IN ORNL CAPSULE EB-2

Type	Density, g/cm ³	Average Grain Size, μm	Total Impurities, w/o ^a	Major Impurities, w/o ^a
Coors AD-995	3.86	13	0.45	Mg, Si, Fe 0.125, 0.07, 0.06 (Cr, 0.125) (Cu, 0.04)
WESGO Al-995	3.85	23	0.28	0.125, 0.1, 0.03
GE Opaque Lucalox	3.91	6	0.06	0.02, 0.007, 0.01 (Ni, 0.01)

^aAverage of results for four samples of each material.

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Volume swelling of Lucalox was considerably less than for either of the other materials and is consistent with previous experimental results¹ for small-grain Lucalox, whereas the swelling for Al-995 is somewhat greater than has been observed previously. The solid cylinders swelled more than the respective cylindrical shells for each of the materials included.

Several specimens were extremely fragile and, in spite of careful handling, were broken while being measured. Of the five Al-995 cylindrical shells, only one survived the complete set of measurements. One of the AD-995 cylindrical shells was found to be broken at the time of capsule disassembly. The remaining cylindrical shells are intact and will be subjected to leak-tightness measurements.

There was no apparent mechanical damage to any of the solid cylinders except for those made of AD-995. Three of the five AD-995 solid cylinders, one of which is shown in Fig. 2, have radial cracks. All AD-995 cylindrical shells have numerous black spots; one of these shells is shown in Fig. 3.

The AD-995 specimens apparently swelled asymmetrically. The ratio of the fractional increase

in length to the fractional increase in diameter for these specimens was ~ 0.65 and was fairly consistent for both the cylindrical shells and the solid cylinders. This ratio was near unity for all other specimens.

It has been noted by Keilholtz, et al² that gross damage to alumina specimens appears to be more a function of impurity content than of grain size. The samples described here would appear to verify this conclusion in that one of the five AD-995 cylindrical shells was broken and three of the five Coors AD-995 solid cylinders had radial cracks, whereas none of the WESGO Al-995 specimens were cracked. As shown in Table IV, the AD-995 had 1.6 times the impurity content and 0.6 the grain size of the Al-995. However, volume increase and post-test fragility were more pronounced for the Al-995 specimens, indicating that grain size was the controlling factor for these parameters. This is consistent with the theory of Wilks, et al³ which predicts that the stored strain energy per unit area of grain boundary increases proportionately to grain size for irradiated alumina. Why such a pronounced difference should exist between the gross-damage results and the swelling and strength results is

Fig. 2. Postirradiation appearance of Coors, AD-995, alumina cylinder.

Fig. 3. Postirradiation appearance of Coors, AD-995, cylindrical-shell alumina insulator. Note black spots.

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still under investigation. Differences in manufacturing technique, as suggested by the anisotropic swelling obtained for AD-999 specimens, and possible thermal-conductivity differences may be influential.

References

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2. G. W. Keilholtz, R. E. Moore, and H. E. Robertson, "Effects of Fast Neutrons on Polycrystalline Alumina and Other Electrical Insulators at Temperatures from 60° to 1230°C," Oak Ridge National Laboratory Report ORNL-4678, May 1971.
3. R. S. Wilks, J. A. Desport, and J. A. G. Smith, "The Irradiation-Induced Macroscopic Growth of α -Al₂O₃ Single Crystals," J. Nuc. Mat. 24, 80-86 (1967).

Status of Irradiation Capsules K-3, K-4, K-5, and K-6

Ceramic and ceramic-metal multilayer Test Capsules K-3 and K-4 were loaded into reactor Grid Position 7C1 of the EBR-II on September 19, 1971, for irradiation beginning with Run 51C. These units contain 87 cylindrical-shell ceramic and ceramic-metal multilayer specimens being irradiated at temperatures of 450°, 600°, and 750°C. A more complete description of the capsules has been given earlier (Quarterly Status Report IA-4747-MS, August 1971). These test units had accumulated a neutron fluence of $\sim 3.8 \times 10^{21}$ nvt ($E_n > 0.1$ MeV) on January 31, 1972. The EBR-II will probably be shut down for an extended period (~ 3.5 months) beginning the first week of April, 1972. It is estimated that by that time IASL's test units will have accumulated a peak neutron fluence of $\sim 6.6 \times 10^{21}$ ($E_n > 0.1$ MeV).

The preliminary design of Test Units K-5 and K-6 has been completed. These isothermal irradiators are designed to test multilayer specimens prototypical of designs applicable to the 1.314-in.-o.d. thermionic fuel rod. Each unit will provide 16 in. of linear space for testing and will be temperature-controlled by a gas-filled heat pipe. The isothermal irradiator portion of one of these capsules has been assembled, and, because the

design is considerably changed from earlier isothermal irradiator units, will be subjected to a bench test to determine its operating characteristics. The heat-pipe test environment will approximate actual reactor conditions much more closely with respect to heat-pipe dump temperature than has been possible in the past. A system is being designed that will utilize oil-cooling at temperatures ranging from 300° to 350°C.

Burnup and Fission-Gas Retention Measurements on SCIP-518, TFE-619, and TFE-621

General

Postirradiation analyses of the General Electric Co. (GE) Test Units SCIP-518, TFE-619, and TFE-621 are nearing completion. In addition to dimensional measurements and metallographic examinations, all fueled emitters were analyzed to determine both the amounts of fission gases retained in the fuel and the average (with respect to irradiation time) fission densities in the fuel.

Fission-gas retention determinations were made by dissolving known quantities of fuel in a mixture of hydrochloric and nitric acids and collecting the evolved gases. The gases were collected in cans each containing 10.0 g of 10X molecular sieve. Gamma-ray emission from the cans was counted by means of a photomultiplier tube and a 100-channel pulse-height analyzer. The counting system was calibrated for the detection of absolute amounts of ⁸⁵Kr. The efficiency of the gas-collecting system was monitored by collecting two successive gas samples from each fuel dissolution and counting each sample.

The procedure used for determining the amounts of retained fission gases was quite satisfactory, as had been the case for previous postirradiation-fuel analyses, except for TFE-621. The irradiation time of TFE-621 was so short (1375 h) and the fractional release of fission gases so large that the fission-product densities were insufficient in two of the emitters to provide high-enough counting rates for good statistical accuracy. The error limits may be as large as $\pm 20\%$ for the top and center emitters of TFE-621.

Fission densities were determined from comparisons of pre- and postirradiation ratios of

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heavy metal atoms in the fuel. Postirradiation values of heavy-metal ratios were obtained from mass-spectrographic analyses. Preirradiation values were obtained from GE. Two essentially independent determinations of fission density were made for each fuel-emitter, one from each of the ratios $^{235}\text{U}/^{238}\text{U}$ and $^{236}\text{U}/^{238}\text{U}$. The results based on the ratio of $^{235}\text{U}/^{238}\text{U}$ depend, very critically, on the original enrichment, whereas (because the original atom fraction of ^{236}U was very small) the results based on the ratio of $^{236}\text{U}/^{238}\text{U}$ are relatively insensitive to the original fuel composition, especially for relatively high burnups.

The fission densities derived from the $^{235}\text{U}/^{238}\text{U}$ measurements were questionable for the SCIP-518 emitter, and were obviously wrong for the emitters from TFE-619 and TFE-621 because negative values were obtained. Previous experience with this method for Mo-UO₂ cermet routinely resulted in agreement within + 1% between fission densities determined by $^{236}\text{U}/^{238}\text{U}$ and $^{235}\text{U}/^{238}\text{U}$ ratios. Thus the original enrichment values are suspect. Unfortunately it is not possible to obtain unirradiated specimens of the fuel used in these emitters, and the original enrichment values therefore cannot be checked. The values reported below are those based on ratios of

$^{236}\text{U}/^{238}\text{U}$ for which errors in initial enrichment are very much less significant.

SCIP-518

Preliminary values of power density and of the fraction of retained fission-product gases were reported for SCIP-518 in the Quarterly Status Report LA-4844-MS, November 1971. The data have been refined, and slightly different results were obtained. Current results are given in Table VI.

TFE-619

The results of fission-gas retention and power-density analyses for TFE-619 are given in Table VII.

TABLE VII
RESULTS OF POWER-DENSITY AND FISSION-GAS-RETENTION MEASUREMENTS FOR TFE-619

	Top Emitter	Center Emitter	Bottom Emitter
Total average fission power, W	960	1010	770
Fission power averaged over emitter interior volume, W/cm ³	102	116	82
Burnup, fissions/cm ³ of full-density fuel x 10 ⁻¹⁹	5.3	5.2	4.0
Fraction of fission gases released from fuel, %	90	90	92

TABLE VI

SCIP-518 OPERATING CONDITIONS

Average input power: 405 W fission plus 42 W γ-heat (447 W total).^a
233 W/cm³ fission averaged over emitter interior volume.

Average fuel burnup: 3.0×10^{20} fissions/cm³ of emitter interior volume.
 5.2×10^{20} fissions/cm³ of full-density fuel volume.

Fraction of fission gases released from the fuel: 93%.^b

Operating Time, h	Output Power (Electrode), W	Efficiency (η), % ^c
0	58.2 (6.8 W/cm ²)	12.8
2 000	45.8	10.0 ^d
4 000	40.3	8.9 ^e
10 178	37.5	8.7

^a470 W predicted.

^bConfirmed to within 1% by measurement of fission-gas pressure in interelectrode space.

^cUncorrected electrode efficiency.

^dVoltage reduction.

^eVoltage increase, fission gas in diode.

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The results for TFE-619 are reasonably consistent. Fission-power levels, predicted by GE to be ~ 700 W per emitter, were exceeded by ~ 40% by the top and center emitters. Because the bottom emitter had tungsten disks interspersed between UO_2 fuel wafers, the fission power in this emitter would be expected to be less than for the other two. The fractions of fission gases released from the fuel are quite large, and the presence of the tungsten disks in the bottom emitter had no apparent significant effect on fission-gas release mechanisms.

TFE-621

The results of fission-gas-retention and power-density analyses for TFE-621 are given in Table VIII.

TABLE VIII

RESULTS OF POWER-DENSITY AND FISSION-GAS-RETENTION MEASUREMENTS FOR TFE-621

	Top Emitter	Center Emitter	Bottom Emitter
Total average fission power, W^a	570	620	650
Fission power averaged over emitter interior volume, W/cm^3	65	71	74
Burnup, fissions/ cm^3 of full-density fuel $\times 10^{-19a}$	1.5	1.6	1.7
Fraction of fission gases released from fuel	98	98	76
Ratio of power density, outside region to inside region	-	2.8	-
Ratio of retained fission gas per unit mass of fuel, outside region to inside region	-	3.1	2.4

^a Based on assumption that each emitter contained the nominal fuel loading of 72 g UO_2 .

Fission power levels for this test were predicted by GE to be ~ 740 W per emitter, which agrees reasonably well with postirradiation-measured values. Because of the apparent unavailability of the actual values of the total amounts of fuel included in these emitters, it has been assumed that each emitter contained the nominal fuel loading of 72 g UO_2 .

The measured fractions of fission gases released from the fuel are very large for the top and center emitters. The values quoted are thought to be correct to within 1%, even though relatively large uncertainties exist regarding the measured fractions

of retained fission gases. These results are consistent with the fact that the test was operated at open-circuit emitter temperatures (~ 2100°C) for several hours at the end of the irradiation period.

The fractional retention of fission gases in the fuel for the bottom emitter is difficult to understand--or even to believe. Metallographic samples of the fuel from this emitter are still available for fission-gas-retention analysis. Once the metallographic examinations have been completed, measurements of the fraction of fission gases retained in the fuel will be repeated.

Fission-gas-retention and fission-power-density measurements were made for fuel samples from different radial positions in the center and bottom emitters of TFE-621. The nominal dimensions of the fuel samples examined and their relative locations are shown in Fig. 4. The results of these measurements are given in the last two lines of Table VIII.

The ratio of the power densities averaged over the two fuel samples at different radial positions indicated in Fig. 4 is equal to 2.8. Because of the relatively large sizes of the samples, the ratio of peak-to-minimum power density was obviously considerably larger than 2.8. The ratio of the amounts of

Fig. 4. Nominal dimensions of fuel samples used to determine radial dependence of fission density and fission-gas retention (all dimensions in inches).

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retained fission gas in the radial samples is only ~ 10% different from the ratio of power densities. Considering the apparent effect of temperature on the fraction of retained fission gases, a greater difference than this was expected.

The analysis of the radial dependence of power density and fission-gas retention was not completely satisfactory. This was due primarily to the difficulty of obtaining fuel samples of precise size and location. Another attempt is being made to obtain such data, and fuel samples have been cut in a reasonably precise manner from the top emitter of TFE-619. A cross-sectional sample of the fuel was potted in epoxy, and fuel samples were cut with an abrasive saw from the potted material. These samples will be depotted and dissolved in preparation for making measurements of isotopic ratios and retention of ^{85}Kr .

Ratio of ^{137}Cs to ^{144}Ce Retained in Fuel

Samples of fuel from SCIP-518 and from the center emitter of TFE-619 have been analyzed for ^{144}Ce and ^{137}Cs . The ratio of ^{137}Cs to ^{144}Ce was found to be 4.52×10^{-3} for SCIP-518 and 4.12×10^{-3} for TFE-619. The ^{137}Cs -vs- ^{144}Ce ratio for fissions in ^{235}U induced by thermal neutrons is known to be 5.21×10^{-2} . Thus, the ^{137}Cs retention relative to that of ^{144}Ce was 9% for SCIP-518 and 8% for TFE-621. The fractional retention of ^{85}Kr was measured to be 7% for SCIP-518 and 10% for the center emitter of TFE-619. It is highly probable that all of the cerium is retained in the fuel. Hence, the fractional retention of cesium in the fuel of these emitters can be said to have been about the same as the fractional retention of gaseous fission products.

Status of ETR Cermet Irradiation Tests

Four test capsules, each containing two 0.5-in.-diam Mo99W-UO₂ cermet fuel specimens are currently being irradiated in the ETR. Irradiation began on September 12, 1971, and 1881 h of irradiation time had been accumulated on January 31, 1972. During a reactor shutdown in early January, three of the four test units were moved to positions of higher neutron flux to compensate for fuel burnup.

Tungsten-Uranium Oxide Cermets

One category of fuel designs of interest for thermionic application consists of configurations that are weak, decoupled mechanically from the cladding, and capable of very high fractional release of fission gases. Mechanically weak, porous W-UO₂ cermets with high UO₂ content are examples. Such cermets could possibly be made by applying thin coatings of tungsten to UO₂ particles that are subsequently pressed and sintered.

A preliminary experiment was conducted to determine the feasibility of applying thin coatings of tungsten to UO₂ particles by means of a fluidized bed. The UO₂ particles ranged in size from about 10 to 25 μm . This is the range of UO₂ particle size observed in porous Mo-UO₂ cermets after extended irradiation. The coating-gas mixture and flow rates were WF₆, 20 ml/min; H₂, 300 ml/min; and Ar, 500 ml/min. The fluidized-bed temperature was 400°C and was operated for 20 min.

A photomicrograph of the resulting product is shown in Fig. 5. The result indicates that UO₂ powders with a particle size as low as 10 μm can be readily coated with thin layers of tungsten.

Fig. 5. Photomicrograph of tungsten-coated UO₂.

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HEAT PIPE SYSTEMS

Introduction

Because of its brevity, the unclassified section on heat pipe systems, normally published separately, has been combined with the remainder of the Space Electric Power R and D Program Quarterly Status Report.

Investigation of Heat Pipe Purification Methods

Recent investigations at IASL indicate that the performance of heat pipes with alkali-metal working fluids can be seriously impaired by impurities collecting in the heated section. As described earlier (see Quarterly Status Report IA-4746-MS, Part I, August 1971) such impairment was eliminated in tests with lithium in Nb-1%Zr heat pipes of the annular-wick geometry by removing oxygen with a hot trap containing yttrium foil. When poor performance was encountered in tests with sodium in stainless-steel heat pipes, a similar technique was used in an attempt to improve performance. Because it is well known that zirconium removes oxygen from sodium effectively, it was used as the getter material. When it was ascertained that the hot-trapping procedure did not yield significant performance improvement, investigations were undertaken to determine if impurities other than oxygen were causing poor performance.

The "annular-wick" geometry, for which impurity problems have been encountered is so-named because the liquid-return path is formed by a gap between an inner tube of fine porosity and an outer container tube. Because this single liquid-flow path lacks the redundancy obtained in earlier heat pipes with wicks consisting of several axial channels covered with screen, it may be very susceptible to impurity problems since even small amounts of impurities could close the pores of the inner tube and cause vapor to form in the annulus. Some evidence of pore closure has been obtained by removing the heated section from a heat pipe with an annular wick after it had been operated with sodium for several hours. Examination of the porous tube from this section has shown that several pores could have been completely closed with copper and other impurities, which have not yet been identified because they exist in complex compounds.

As described earlier (see Quarterly Status Report IA-4844-MS, November 1971) some tests with sodium in stainless-steel heat pipes indicate that the impurities that are causing problems may be removed by cold-trapping. These tests are continuing with two annular-wick heat pipes obtained from Westinghouse Astronuclear Laboratory. Should these tests be successful, it is hoped that the cold traps can be removed and their impurities identified. Such identification is important because it could help establish heat-pipe preparation procedures which might eliminate the source of impurities.

PUBLICATIONS

J. C. Rowley, C. R. Saunders, C. McInteer, C. P. Kempter, and R. E. Riley, "Elastic Moduli and Thermal Expansion of Porous Molybdenum-Rhenium Dioxide Elements," IASL Report IA-4825, November 1971.

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