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THE EFFECT OF NUCLEAR RADIATION ON DISPERSION FUEL MATERIALS

484683

Report No. 40

RADIATION EFFECTS INFORMATION CENTER
Battelle Memorial Institute
Columbus Laboratories
Columbus, Ohio 43201

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REIC Report No. 40
January 1, 1966

REPORT

on

THE EFFECT OF NUCLEAR RADIATION ON DISPERSION FUEL MATERIALS

by

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to

AERONAUTICAL SYSTEMS DIVISION
AIR FORCE SYSTEMS COMMAND
WRIGHT-PATTERSON AIR FORCE BASE

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ABSTRACT

A complete summary of the irradiation performance of the various fuel materials dispersed in different matrices has been collected. The irradiation performance of the following dispersions is discussed in the report: BeO-UO_2, BeO-UO_2-ThO_2, Al_2O_3-UO_2, MgO-UO_2, UO_2, UC, UC_2, and $(U,Th)C_2$ dispersed in graphite; UBe_{13} and $(U,Th)B_{13}$ dispersed in beryllium; UO_2, U_3O_8, U_3O_8 glass, and UAl_3 dispersed in aluminum; UO_2, UC, and UN dispersed in stainless steel; UO_2 dispersed in nickel-chromium alloys; UO_2 dispersed in niobium, UO_2 dispersed in molybdenum, UO_2, UN, and UC dispersed in tungsten. The data which is summarized includes percentage of fuel by weight and volume, specimen shape and size, fuel particle size, cladding material and clad thickness, irradiation temperature, heat generation rates, irradiation times, fuel burnup, fission gas release, dimensional and density changes resulting from irradiation.

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SUMMARY

Ceramic Materials

Beryllium Oxide

Experiments have shown that for best irradiation performance, UO_2 particle size should be around 100 μ. Use of 100-μ UO_2 particles results in better dimensional stability of the fuel during irradiation, although the fission-gas release is increased by increased particle size. Irradiation of the BeO-UO_2 fuel results in a fission-fragment-damaged zone in the BeO surrounding each UO_2 particle. Special etching of the irradiated BeO-UO_2 is required to make this reaction zone distinct, and it has been observed only in dispersions that have been irradiated at low temperatures, to burnups above 10×10^{20} fissions/cm^3. Although in some instances it was not possible to detect a distinct phase, the UO_2 particles "fell out" during metallographic preparation, indicating that they were surrounded by a brittle material that in all probability was the fission-fragment damaged zone. The observance of the damaged zone in only the material irradiated at low temperatures raises the possibility that the fission-fragment damage is annealed out at high irradiation temperatures. Other irradiation-induced microstructural changes that have been observed are: fission-gas-bubble formation in the UO_2 particles and an increase of size of the UO_2 particles. Both of these are favored by high irradiation temperatures. From the irradiation experiments it seems that BeO-UO_2 fuel pins can be irradiated at surface temperatures up to 900 C for burnups of 7×10^{20} fissions/cm^3 if a Hastelloy X cladding is used. Use of stainless steel as a cladding material reduces the irradiation performance somewhat.

The irradiation performance of BeO-UO_2-ThO_2 fuel pins seems to be good up to the irradiation conditions tested. As of the present, the specimens have been irradiated at surface temperatures of 700 C, with the maximum burnup being 2×10^{20} fissions/cm^3. These specimens have retained their dimensional stability and have also exhibited good fission-gas-retention properties.

Aluminum Oxide

Fuel elements utilizing Al_2O_3-UO_2 dispersions have shown very poor dimensional stability when irradiated at maximum central temperatures of 740 C and burnups of only 0.9×10^{20} fissions/cm^3. Irradiation of similar specimens to burnups of 5.2×10^{20} fissions/cm^3 did not result in further volume increases. These tests indicate that the performance of Al_2O_3-UO_2 dispersions is considerably inferior to performance of BeO-UO_2 dispersions.

Magnesium Oxide

The limited irradiation tests carried out on MgO-UO_2 fuel elements are inconclusive since the low-density material had an expected high fission-gas release. Further tests would be required to evaluate the fuel completely; however, it seems unlikely that the performance of MgO-UO_2 dispersions would equal the excellent performance of BeO-UO_2 dispersions.

Graphite

Graphite-matrix dispersions showed good irradiation performance at surface temperatures of 800 C and burnups of up to 0.8×10^{20} fissions/cm^3. However, precoating the fuel particles with various coatings has improved the fission-gas retention of the dispersions to such an extent that no further interest has been shown in graphite dispersions that utilize uncoated particles.

Metallic Materials

Beryllium

Irradiation experiments have been performed on hot-pressed specimens containing particles of (U,Th)Be_{13} and UBe_{13} dispersed in a beryllium matrix. The specimens have been irradiated at surface temperatures of 435 to 530 C and burnups up to 1.1×10^{20} fissions/cm^3. These specimens underwent a maximum density decrease of 5.1 percent, the maximum fission-gas release being 1.8 percent. Irradiation of similar specimens, at about the same irradiation temperatures, to burnups above 3×10^{20} fissions/cm^3 resulted in volume increases of 11 to 18 percent, with the fission gas release being above 10 percent. Specimens that contained fuel particles of <50 μ released somewhat smaller quantities of fission gases but underwent larger dimensional changes than specimens containing 100- to 200-μ particles. No microstructural changes resulted from the irradiation.

Aluminum

Considerable numbers of irradiations has shown that fuel elements that contain uranium oxides dispersed in aluminum are stable when irradiated at 50 to 60 C. However, irradiations of UO_2 and U_3O_8 particles contained in aluminum were found to be stable only to a burnup of 6 to 7×10^{20} fissions/cm^3 when irradiated at about 200 C. The limiting factor appeared to be the reaction between the uranium oxides and aluminum at that temperature which caused swelling due to production of low-density compounds. This reaction between aluminum and uranium oxides seems to be enhanced by irradiation.

Irradiations of UAl_3 particles dispersed in aluminum showed that these fuel elements were stable when irradiated to burnups of 11×10^{20} fissions/cm^3. Further irradiations to burnups of 35×10^{20} fissions/cm^3 resulted in volume increases of 5.4 to 6.1 percent, almost all of the volume change being due to thickness changes.

Specimens containing U_3O_8 glass dispersed in an aluminum matrix showed fairly good dimensional stability when irradiated to 3.7×10^{20} fissions/cm^3 at 100 C.

Iron-Base Alloys

Specimens containing UO_2 particles dispersed in an austenitic stainless steel matrix have been thoroughly investigated. As a result of improvements in fabrication techniques, the irradiation performance with respect to temperature and burnup has

been continuously increased. The present status of irradiation performance of stainless steel-UO_2 dispersions is shown in Figure 15. Figure 15 shows that specimens can be irradiated at 900 C to a burnup of 3×10^{20} fissions/cm^3 or at 350 C to a burnup of 24×10^{20} fissions/cm^3 without failure or excessive swelling. The limiting factor in the performance of these dispersions is the swelling of UO_2 due to fission-product accumulation with increasing burnup. As the temperature of the matrix is increased the yield strength of the matrix decreases, and swelling becomes more pronounced if the burnup stays the same. Another important factor that determines irradiation performance is the bonding between the matrix and the cladding. If the bonding is not complete, local overheating will take place, which in turn results in blistering. The main microstructural changes that are caused in the dispersions by irradiation are bubble formation in the UO_2 particles, formation of an unidentified white metallic phase in UO_2 particles, a fission-fragment-damaged zone in the stainless steel that surrounds the UO_2 particles, and cracking in both the stainless steel matrix and UO_2 fuel particles.

The performance of UC, UN, and PuO_2 dispersed in stainless steel is very similar to the performance of UO_2 except that incompatibility between the stainless steel matrix and UC and UN has been detected at high operating temperatures.

Refractory Metals

A limited number of irradiations involving fuels dispersed in refractory metals have been reported in unclassified literature. UO_2 and UN dispersed in molybdenum and tungsten have been irradiated to burnups of $0.2 - 1.2 \times 10^{20}$ fissions/cm^3 with the irradiation temperatures being 2000-2400 C. The specimens did not undergo dimensional changes, although they released a large percentage of their fission gases.

INTRODUCTION

The reason for developing dispersion fuels was to improve the irradiation performance of nuclear fuels. The two earliest nuclear fuels considered – metallic uranium and uranium dioxide – proved to be unsatisfactory for applications requiring moderate surface temperatures and fuel stability after irradiation. Uranium dioxide was deemed unacceptable because of its low thermal conductivity, which caused central melting of the fuel if a specimen of moderate size was operating at a moderate heat-generation rate. Uranium metal was found to be unsatisfactory because of its dimensional instability as a result of irradiation at moderate temperatures.

A dispersion fuel can be made to vary in its performance by choosing the matrix and fuel material for the required application. The main improvements that usually can be obtained by using a matrix material are improved thermal conductivity, which results in lower operating temperatures, and higher strength, which results in improved dimensional stability during irradiation. In special cases the matrix material can also be chosen to act as a moderating material for the reactor. In selecting a matrix material, it is absolutely necessary that the matrix material and the fuel are compatible at the maximum operating temperatures of the fuel element. If they are not compatible, reaction products are formed which may have different densities and thereby cause dimensional changes in the dispersion. The reaction may also result in failure of the matrix, thus enabling the coolant to come into contact with the fuel material. The linear thermal-expansion coefficient of the fuel should be less than that of fuel. If the linear thermal expansion of the fuel is greater, spalling and erosion of the fuel particle would be expected as a result of thermal cycling. Also, the whole dispersion may grow as a result of thermal cycling when the fuel expands more than the matrix and thereby causes a plastic deformation of the matrix material. The main disadvantage of dispersion fuels is that the matrix material acts as a diluent in the system requiring either a larger reactor core or a higher enrichment of fertile material to obtain criticality. Also, a matrix material with a small capture cross section for neutrons should be considered in order to promote neutron economy.

In selecting a fuel material, the object is to select the fuel with the highest uranium content (UC and UN rather than UO_2) in order to reduce the core size required for criticality. In all of these choices, the overall cost of the materials and fabrication should be considered in making the decision.

In designing a dispersion it becomes important to consider the size of the fuel particles in the matrix. An important consideration is to minimize the fraction of the matrix material that is damaged by fission products. Very little experimental work is available for the ranges of recoiling fission fragments in various materials, but a value of 10 μ is generally assumed. The recoil range has been found to decrease with increasing density. It has been found that fission-product bombardment decreases the ductility of the matrix material. Therefore it becomes necessary to have a continuous matrix that is undamaged by fission fragments, in order to have ductile material available for plastic deformation. The ductility is used to accommodate the pressure build-up caused by volume expansion of fuel particles caused by fission-product accumulation. Figure 1 shows the percentage of the matrix that is expected to be damaged by recoiling fission fragments at different fuel loadings and fuel-particle sizes. This

requirement dictates that the fuel particles should not be too small in diameter, as that would result in fission-fragment damage to the whole matrix. On the other hand, the diameter of the fuel particles should not be so large as to cause high temperatures in the fuel particles with poor thermal conductivity. Therefore, the particle size of fuel becomes a compromise depending on the matrix material, fuel material, and the operating conditions (surface temperature and heat-generation rate) of the proposed dispersion fuel element. Experience has shown that the best size for fuel particles is about 100 μ. Another factor to consider is the damage caused by neutrons to the matrix material. The general mode of damage is the displacement of atoms from their lattice sites. It has been determined that a neutron with an energy of at least 30 ev is required for displacement of an atom from its lattice site.[1] However, neutrons having greater energies will cause a larger number of displacements by knocking the displaced atoms into other atoms, which in turn displace other atoms, until all the kinetic energy contained by the original neutron is finally dissipated. It has been found that the majority of atoms that are displaced in materials are displaced by neutrons having energies greater than 1 Mev. Displacement of atoms in metals by fast neutrons causes an increase in ultimate strength and a more pronounced increase in yield strength, while the ductility is drastically decreased. In ceramics, the displacement of atoms by fast neutrons causes a decrease in compressive and tensile strengths of most materials.

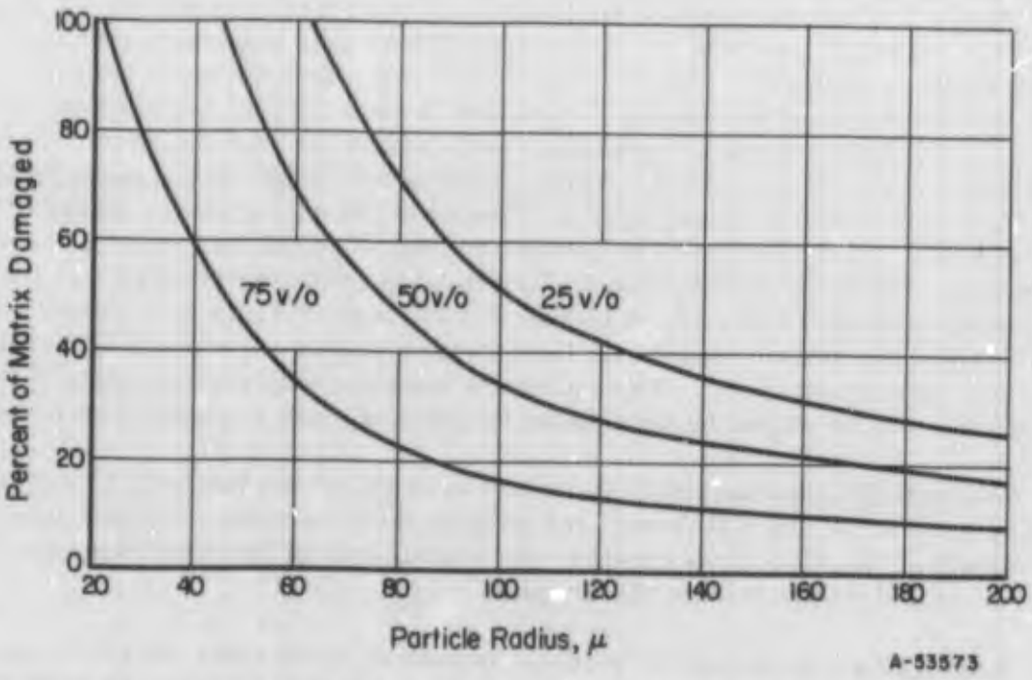


FIGURE 1. MATRIX DAMAGE AS A FUNCTION OF FUEL VOLUME AND FUEL-PARTICLE SIZE

Fission-fragment range = 10 μ; spherical fuel particles.

Since the displaced atoms are knocked into an unstable position, they will have a tendency to return to their original stable positions by diffusion, provided that the temperature is high enough. Therefore, an equilibrium number of displacements is established in the material which is being exposed to fast neutrons. This equilibrium number of displacements depends on rate of fast-flux accumulation, temperature of material, and the material itself. Therefore, each material has a specific temperature above which no appreciable fast-neutron damage takes place when the material is

being irradiated at that temperature. However, this temperature is usually sufficiently high so that the strength of the material is considerably reduced when operated at that temperature. It has also been found that a higher annealing temperature is required to remove the damage caused by fission fragments than is required to remove the damage caused by fast neutrons.[2] This may be due to the fission fragments acting as impurity atoms. Therefore, a compromise has to be made as to the amount of fast-neutron damage that can be accepted and the operating temperature of the dispersion.

Another important consideration in the performance of the dispersion is the process by which it was fabricated. The earlier dispersions exhibited angular fuel particles that would result in stringers during the rolling operation. Stringers in turn would act as stress raisers where cracks would originate. Adoption of more spherical particles and an increase of the rolling temperature has eliminated this problem. To insure a continuous matrix in highly loaded dispersions, it has become necessary to precoat the fuel particles with a thin coat of matrix material before sintering.

Some of the pertinent properties of important matrix and fuel materials are given in the Appendix. The properties that are given are compatibility, physical properties, and the annealing temperature required to remove fast-neutron damage.

CERAMIC MATRIX MATERIALS

Beryllium Oxide

Beryllium oxide has found wide application as a matrix material for UO_2 dispersions because of its many good properties. Among these may be included its rather good thermal conductivity, resistance to thermal shock, low absorption of thermal neutrons, and good mechanical properties at elevated temperatures. Although an eutectic is formed between BeO and UO_2 at 2050 C and 65 molecular percent BeO, no reaction takes place between the two at the expected operating temperatures of the dispersion. Fast neutrons cause expansion of BeO lattice but this becomes negligible at higher irradiation temperatures. The production and release of helium by (n,α) reaction in BeO has also been found to be negligible. Another advantageous property of BeO fuel is its neutron-moderating ability. Thus a BeO-dispersed fuel element can be used as a self-moderating, homogeneous reactor fuel.

The first irradiation studies of a BeO-UO_2-dispersed fuel were carried out at Argonne National Laboratory.[3] Cylinders and prisms of BeO containing 2 and 10 weight percent UO_2 were sintered to densities ranging from 70 to 90 percent of theoretical density. The UO_2 particle size was less than 2 microns, while the BeO grain size was from 6 to 12 microns. The samples containing 2 weight percent UO_2 were irradiated at 250 C, while the 10 weight percent UO_2 specimens were irradiated at 650 to 700 C. Burnup ranges of these specimens were from 5.5×10^{17} to 1.2×10^{19} fissions/cm^3. All of the BeO-UO_2 specimens underwent linear expansions of about 0.5 to 0.8 percent. Performance of all significant BeO-UO_2 irradiations is given in Table 1. It was felt that the reason for the somewhat uniform expansion at the two

TABLE 1. IRRADIATION PERFORMANCE OF UO_2 FUEL

Specimen Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Density, % T.D.(a)	Specimen Shape	Specimen Length, in.	Specimen Diam or Thickness, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.
ANL	1	2	0.5	76	Rod	0.25	0.25	~2	Al	
	1	2	0.5	76	Rod	0.25	0 25	2	Al	
	1	2	0.5	76	Rod	0.25	0.25	2	Al	
	1	2	0.5	82	Rod	0.25	0.25	2	Al	
	1	2	0.5	82	Rod	0.25	0.25	2	Al	
	1	2	0.5	82	Rod	0.25	0.25	2	Al	
	1	10	3.0	76	Rod	0.25	0.25	2	Al	
	1	10	3.0	76	Rod	0.25	0.25	2	Al	
	1	10	3.0	76	Rod	0.25	0.25	2	Al	
	1	10	3.0	79	Rod	0.25	0.25	2	Al	
	1	10	3.0	79	Rod	0.25	0.25	2	Al	
	1	10	3.0	79	Rod	0.25	0.25	2	Al	
	1	2	0.5	88	Plate	1.75	0.19 x 0.19	2	Al	
	1	2	0.5	88	Plate	1.75	0.19 x 0.19	2	Al	
	1	2	0.5	88	Plate	1.75	0.19 x 0.19	2	Al	
	1	2	0.5	89	Plate	1.75	0.19 x 0.19	2	Al	
	1	2	0.5	89	Plate	1.75	0.19 x 0.19	2	Al	
	1	2	0.5	89	Plate	1.75	0.19 x 0.19	2	Al	
	1	10	3.0	74	Plate	1.75	0.19 x 0.19	2	Al	
	1	10	3.0	74	Plate	1.75	0.19 x 0.19	2	Al	
	1	10	3.0	74	Plate	1.75	0.19 x 0.19	2	Al	
	1	10	3.0	79	Plate	1.75	0.19 x 0.19	2	Al	
	1	10	3.0	79	Plate	1.75	0.19 x 0.19	2	Al	
	1	10	3.0	79	Plate	1.75	0.19 x 0.19	2	Al	
1-A	1	25.3	8.5	96	Plate	1.5	0.25 x 0.04	<50	Zr-2	0.016
B	1	25.3	8.5	96	Plate	1.5	0.25 x 0.04	<50	Zr-2	0.016
C	1	25.3	8.5	96	Plate	1.5	0.25 x 0.04	<50	Zr-2	0.016
D	1	25.3	8.5	96	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
2-A	1	28.7	10	90	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
B	1	28.7	10	90	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
C	1	28.7	10	90	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
D	1	28.7	10	90	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
3-A	1	30.1	10.5	91	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
B	1	30.1	10.5	91	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
C	1	30.1	10.5	91	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
D	1	30.1	10.5	91	Plate	1.5	0.25 x 0.04	150	Zr-2	0.016
29-22	1	65	34	97	Plate	1.5	0.25 x 0.03	10	Zr-2	0.015
59-2	1	65	34	97	Plate	1.5	0.25 x 0.02	10	Zr-2	0.020
59-3	1	51	22	97	Plate	1.5	0.25 x 0.03	10	Zr-2	0.018
1B-8T-1	12	60	29	94-97	Rod	28	0.180		Hastalloy-X	0.030
1B-8T-2	12	60	29	94-97	Rod	28	0.180		Hastalloy-X	0.030
MTR-GCR-3	1	70	39	96-97	Rod	1.056	0.145	5-40	Hastalloy-X	0.030
11	1	80	53	96-97	Rod	1.056	0.145	5-40	Hastalloy-X	0.030
4	1	70	39	96-97	Rod	1.056	0.145	5-40	Hastalloy-X	0.030
6	1	70	39	96-97	Rod	1.056	0.145	5-40	Hastalloy-X	0.030
13	1	70	39	96-97	Rod	1.056	0.145	5-40	Hastalloy-X	0.030
10	1	70	39	96-97	Rod	1.056	0.145	5-40	Hastalloy-X	0.030
1B-17R-2	1	75	45		Rod		0.180	100-500	Hastalloy-X	0.033
3	1	75	45		Rod		0.180	100-500	Hastalloy-X	0.033

PARTICLES DISPERSED IN BERYLLIUM OXIDE MATRIX

Temperature, C			Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3 of fuel	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %				
Surface	Average	Center						Diam or Thickness	Density	Volume	Length,	Reference
	250			0.006							0.50	3
	250			0.013							0.90	3
	250			0.022							0.60	3
	250			0.006							0.62	3
	250			0.014							0.52	3
	250			0.024							0.65	3
	650-700			0.034							0.44	3
	650-700			0.069							0.71	3
	650-700			0.11							0.62	3
	650-700			0.035							0.61	3
	650-700			0.071							0.46	3
	650-700			0.12							0.65	3
	250			0.006							0.51	3
	250			0.015							0.60	3
	250			0.026							0.59	3
	250			0.006							0.48	3
	250			0.015							0.67	3
	250			0.026							0.65	3
	650-700			0.032							0.61	3
	650-700			0.066							0.73	3
	650-700			0.11							0.78	3
	650-700			0.034							0.60	3
	650-700			0.070							0.64	3
	650-700			0.12							0.81	3
280	475			11.1			19.9	27				4
280	475			11.1				30				4
280	475			11.1				28.8				4
280	475			11.1				27.5				4
280	500			12.1			6.9	3.0				4
280	500			12.1				8.0				4
280	500			12.1				4.5				4
280	500			12.1				7.3				4
280	445			10.7				10.7				4
280	445			10.7				12.7				4
280	445			10.7				27.5				4
280	445			10.7				8.2				4
235		690		10.8	2.9	193	0.13			5.5		5
235		860		35.7	3.4	224	5.1			2.9		5
235		1130		28.0	4.5	295	2.6			2.4		5
955			6415	3.3	0.45	51	0.2-0.7					6
955			10150	5.7	0.49	55	0.1-0.8					6
855		1400	8750	5.2	0.52	48	2.8	0.5			0.2	8
930		1365	8750	5.0	0.50	46	69	15 Ruptured			0.1	8
		1660	8750	6.3	0.63	58	1.3	0.5			0.1	8
870		1685	8750	6.4	0.64	59	1.5	0.1			0.1	8
860		1570	8750	5.9	0.59	54	2.8	0.1			0.1	8
840		1545	8750	5.8	0.58	53	0.6	-0.5			0.1	8
770		910	5820	3.1	0.46	52		No dimensional changes				9
715		825	5820	2.6	0.39	45	1.4	No dimensional changes				9

TABLE I.

Specimen Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Density, % T.D.(a)	Specimen Shape	Specimen Length, in.	Specimen Diam or Thickness, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.
1B-17R-6	1	75	45		Rod		0.180	100-500	Hastalloy-X	0.033
10	1	75	45		Rod		0.180	100-500	Hastalloy-X	0.033
12	1	75	45		Rod		0.180	100-500	Hastalloy-X	0.033
18	1	75	45		Rod		0.180	100-500	Hastalloy-X	0.033
BMI-47-1										
7	1	70	39	92.3	Rod	0.75	0.140		Hastalloy-X	0.030
8	1	70	39	91.7	Rod	0.75	0.140		Hastalloy-X	0.030
9	1	70	39	91.9	Rod	0.75	0.140		Hastalloy-X	0.030
MGCR-4	4	60	29		Rod	2.0	0.390	25	Monel	0.015
	4	60	29		Rod	2.0	0.390	150	Monel	0.015
MGCR-1	2	60	29		Rod	18	0.250	150	Nickel	0.010
2		60	29		Rod	18		1	Monel	0.010
3	19	60	29		Rod		0.375		Hastalloy-X	0.010
GCRE-BRR-II-1	1	37-56	14-26	98	Rod	1.01	0.16	<40	Hastalloy-X	0.030
2	1	37-56	14-26	98	Rod	1.01	0.16	<40	Hastalloy-X	0.030
3	1	37-56	14-26	98	Rod	1.01	0.16	<40	Hastalloy-X	0.030
4	1	37-56	14-26	98	Rod	1.01	0.16	<40	Hastalloy-X	0.030
5	1	37-56	14-26	98	Rod	1.01	0.16	<40	Hastalloy-X	0.030
6	1	37-56	14-26	98	Rod	1.01	0.16	<40	Hastalloy-X	0.030
BMI-31-1										
1	1	54	24		Rod	0.25	0.22	125	316 SS	0.011
2	1	54	24		Rod	0.25	0.22	125	316 SS	0.011
3	1	54	24		Rod	0.25	0.22	125	316 SS	0.011
4	1	54	24		Rod	0.25	0.22	125	316 SS	0.011
BMI-31-3										
6	1	47	19	94.5	Rod	0.25	0.22	125	316 SS	0.011
7	1	47	19	94	Rod	0.25	0.22	125	316 SS	0.011
10	1	47	19	93.8	Rod	0.25	0.22	125	316 SS	0.011
12	1	47	19	93.8	Rod	0.25	0.22	125	316 SS	0.011
13	1	47	19	95.2	Rod	0.25	0.22	125	316 SS	0.011
14	1	47	19	93.1	Rod	0.25	0.22	125	316 SS	0.011
17	1	47	19	93.1	Rod	0.25	0.22	125	316 SS	0.011
24	1	47	19	93.1	Rod	0.25	0.22	125	316 SS	0.011
BMI-31-2										
8	1	52	22.5	89.6	Rod	0.25	0.22	125	316 SS	0.011
9	1	52	22.5	89.1	Rod	0.25	0.22	125	316 SS	0.011
12	1	52	22.5	89.3	Rod	0.25	0.22	125	316 SS	0.011
17	1	52	22.5	89.8	Rod	0.25	0.22	125	316 SS	0.011
2	1	52	20.5	93.8	Rod	0.25	0.22	125	316 SS	0.011
8	1	52	20.5	94.8	Rod	0.25	0.22	125	316 SS	0.011
21	1	49	20.5	94.0	Rod	0.25	0.22	125	316 SS	0.011
22	1	49	20.5	93.8	Rod	0.25	0.22	125	316 SS	0.011
BMI-31-4										
5	1	52	22.5	90.5	Rod	0.25	0.22	125	316 SS	0.011
13	1	52	22.5	90.8	Rod	0.25	0.22	125	316 SS	0.011
30	1	52	22.5	90.8	Rod	0.25	0.22	125	316 SS	0.011
31	1	52	22.5	89.4	Rod	0.25	0.22	125	316 SS	0.011
3	1	52	20.5	95.1	Rod	0.25	0.22	125	316 SS	0.011

(CONTINUED)

Temperature, C			Irrad.	Burnup, 10^{20} fissions/ cm^3 of	Bulk Heat Gen.	Surface Heat Gen.	Fission-Gas	Property Changes Caused by Irradiation, %				
Surface	Average	Center	Time, hr	fuel	Rate, kw/cm^3	Rate, w/cm^2	Release, %	Diam or Thickness	Density	Volume	Length,	Reference
630		680	5820	1.7	0.25	28	1.0		No dimensional changes			9
715		825	5820	2.8	0.42	48			No dimensional changes			9
770		910	5820	3.3	0.49	56	2.2		No dimensional changes			9
730		815	5820	1.9	0.28	32	1.3		No dimensional changes			9
940		1120	4700	14.8	2.7	236		5.2	-10.0		0	10
930		1090	4700	14.5	2.7	236		6.4	-11.1		4.3	10
800		970	4700	12.4	2.3	200	13.4	3.3	-8.4		0.	10
		1095	11700	3.7	0.28	70	0.1	0.4-0.8		1.5-3.0	0.2-0.5	11
		1095	11700	3.7	0.28	70						11
820			1000						No dimensional changes			12
820			1300						No dimensional changes			12
930			1430						No dimensional changes			12
			1200	1.1	0.80	77						13
930			1200	1.4	1.0	97						13
940			1200	1.2	0.87	84						13
900			1200	1.1	0.80	77			-1.1			13
890			1200	0.83	0.60	58						13
790			1200	0.89	0.65	63			-1.6			13
815	955-1025		1150	0.87	0.66	92		0				14,15,16
815	955-1025		1150	0.87	0.66	92	0.4	0				14,15,16
815	955-1025		1150	0.87	0.66	92		0				14,15,16
815	955-1025		1150	0.87	0.66	92		0				14,15,16
815	955-1025		5400	2.0	0.32	45		0.2			1.0	14,15,16
815	955-1025		5400	2.0	0.32	45		0			0.5	14,15,16
815	955-1025		5400	2.0	0.32	45		0.2			1.0	14,15,16
815	955-1025		5400	2.0	0.32	45		0.1			-0.1	14,15,16
815	955-1025		5400	2.0	0.32	45		-1.0	-1.0		0.	14,15,16
815	955-1025		5400	2.0	0.32	45		-1.0	-0.5		0.4	14,15,16
815	955-1025		5400	2.0	0.32	45		-0.4	0.2		0.2	14,15,16
815	955-1025		5400	2.0	0.32	45		-0.2	0.2		0	14,15,16
815	955-1025		7100	2.6	0.32	45		2.2	+2.5		-2.8	14,15,16
815	955-1025		7100	2.6	0.32	45		1.3	-0.5		2.4	14,15,16
815	955-1025		7100	2.6	0.32	45		1.8	-1.1		1.2	14,15,16
815	955-1025		7100	2.6	0.32	45		1.8	-3.2		1.2	14,15,16
815	955-1025		7100	2.7	0.33	46		-0.4	-0.9		3.0	14,15,16
815	955-1025		7100	2.7	0.33	46		-0.6	-1.1		-0.5	14,15,16
815	955-1025		7100	2.7	0.33	46		-0.6	-1.4		-0.4	14,15,16
815	955-1025		7100	2.7	0.33	46		-0.4	0.9		1.0	14,15,16
815	955-1025		10300	5.8	0.49	69						14,15,16
815	955-1025		10300	5.8	0.49	69						14,15,16
815	955-1025		10300	5.8	0.49	69						14,15,16
815	955-1025		10300	5.8	0.49	69		2.5	-5.5		1.2	14,15,16
815	955-1025		10300	5.1	0.43	60		0.8	-5.4		2.2	14,15,16

TABLE 1.

Specimen Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume %	Density, % T.D.(a)	Specimen Shape	Specimen Length, in.	Specimen Diam or Thickness, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.
BMI-31-5	1	49	20.5	95	Rod	0.25	0.22	125	316 SS	0.011
16	1	49	20.5	92.5	Rod	0.25	0.22	125	316 SS	0.011
19	1	49	20.5	93	Rod	0.25	0.22	125	316 SS	0.011
MGCR-9										
1	1	60	29	95.2	Rod	1.0	0.40	25	Hastalloy-X	
2	1	60	29	95.4	Rod	1.0	0.40	25	Hastalloy-X	
3	1	60	29	93.2	Rod	1.0	0.40	150	Hastalloy-X	
4	1	60	29	94.7	Rod	1.0	0.40	25	Hastalloy-X	
5	1	60	29	93.2	Rod	1.0	0.40	150	Hastalloy-X	
6	1	60	29	93.4	Rod	1.0	0.40	150	Hastalloy-X	
LM-2a	1	59	28	98	Tubular			0.5-5	Inconel	
LM-2b	1	59	28	98	Tubular			0.5-5	Inconel	
LM-4a	1	59	28	98	Tubular			0.5-5	Inconel	
LM-4b	1	59	28	98	Tubular			0.5-5	Inconel	
01-6	4	62	30.5	95	Tubular	5.5	0.500 OD 0.250 ID	150	S.S.	0.20

(CONTINUED)

Temperature, C			Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3 of fuel	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %				
Surface	Average	Center						Diam of Thickness	Density	Volume	Length,	Reference
815	955-1025		10300	5.1	0.43	60		0.4	-5.4		2.2	14,15,16
815	955-1025		10300	5.1	0.43	60					0.9	14,15,16
815	955-1025		10300	5.1	0.43	60		1.7	-2.6		1.0	14,15,16
930				2.2			0		-3.4			17
930				2.1			0.3					17
930				2.3			24.2		-4.5			17
930				2.0			0		-3.5			17
930				2.			23.2		-2.4			17
930				2.3			17.6					17
	1010			3.2			0.3	0.9	-6.3		1.7	21
		1200		3.2			0.3	2.0	8.7		2.5	21
		1200		3.7			0.2	0.9	-5.2		1.2	21
		1200		3.7			0.2	Capsule leaked				21
370		1160		1.1	0.45	76			-1			22,23

different temperatures was the offsetting conditions of low irradiation temperature and low burnup in one set of specimens and high burnup and high irradiation temperature in the other set of specimens. At the higher temperature, a considerable amount of the damage is expected to be annealed out. It was also found that the crushing strength of irradiated specimens was decreased 20 to 30 percent if the density of specimens was more than 80 percent of theoretical density. Increases of up to 30 percent in crushing strength took place in irradiated specimens that had a theoretical density of less than 80 percent. It was also found that the elastic modulus of $BeO-UO_2$ was decreased by irradiation. This is illustrated in Figure 2. The decrease in elastic modulus is

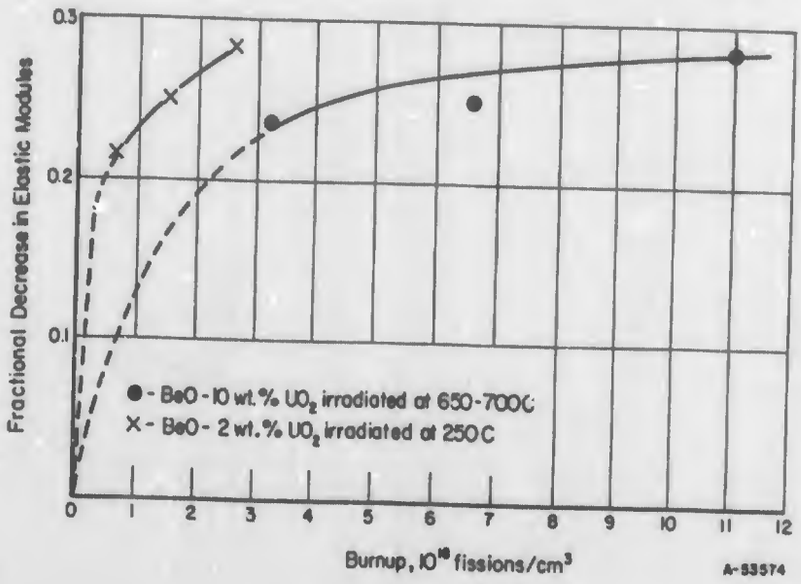


FIGURE 2. EFFECT OF BURNUP ON ELASTIC MODULUS OF $BeO-UO_2$[3]

believed to reach a saturation point after an exposure of about 3×10^{18} fissions/cm^3. Again, the higher temperature anneals out some of the fission fragment damage. The thermal resistivity of the $BeO-UO_2$ fuel was found to increase considerably by neutron exposure, as is shown in Figures 3 and 4. In comparing the effect of irradiation on BeO and $BeO-UO_2$, it can be seen that the majority of the increase is due to fission-fragment effects rather than fast neutrons. The small increase in thermal resistivity at higher burnups is attributed to annealing effects at the higher irradiation temperatures.

Irradiations of sintered $BeO-UO_2$ plates have been performed at Westinghouse.[4,5] These irradiations have been performed on both fine (10 μ) and coarse (150 μ) UO_2 particles. Irradiation temperatures of these specimens were in the 450 to 500 C range, while the burnup was about 10 to 12×10^{20} fissions/cm^3. The specimens containing the fine UO_2 particles underwent considerably larger volume increases than the specimens containing large fuel particles when both were irradiated at the same temperature and burnup. This fact seems to indicate that it is important to keep the fraction of the BeO matrix, which is damaged by fission fragments, to a minimum. In another experiment it was shown that the volume increases undergone by specimens containing fine UO_2

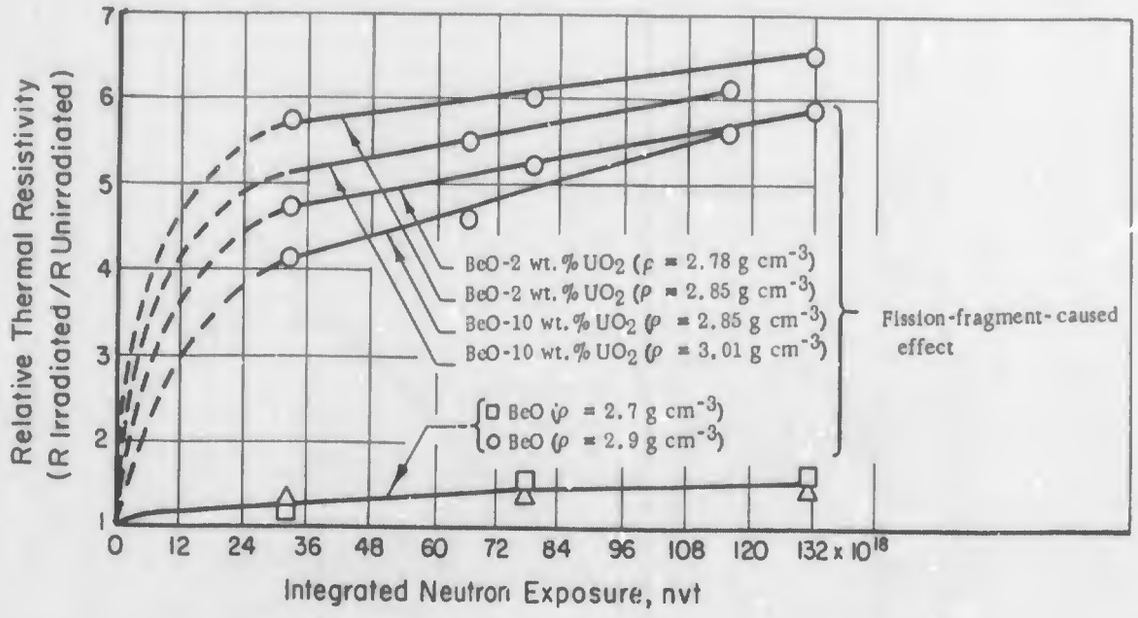


FIGURE 3. EFFECT OF NEUTRON EXPOSURE ON THE RELATIVE THERMAL RESISTIVITY OF BeO AND BeO-UO_2[3]

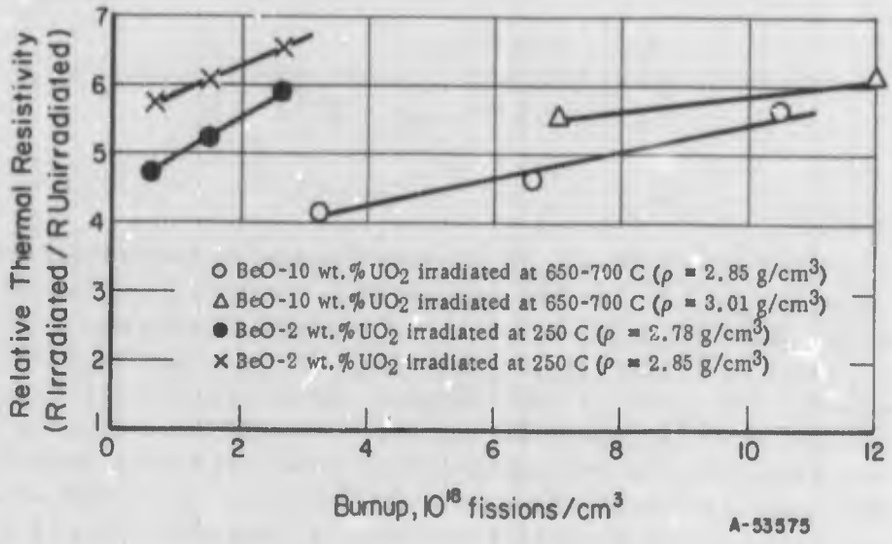


FIGURE 4. EFFECT OF BURNUP AND TEMPERATURE ON THE THERMAL RESISTIVITY OF BeO-UO_2[3]

particles were linearly dependent on burnup (Figure 5). However, the specimens that received the higher burnup also operated at higher temperatures. Therefore, caution must be exercised in assuming that the magnitude of volume increase is due only to magnitude of burnup.

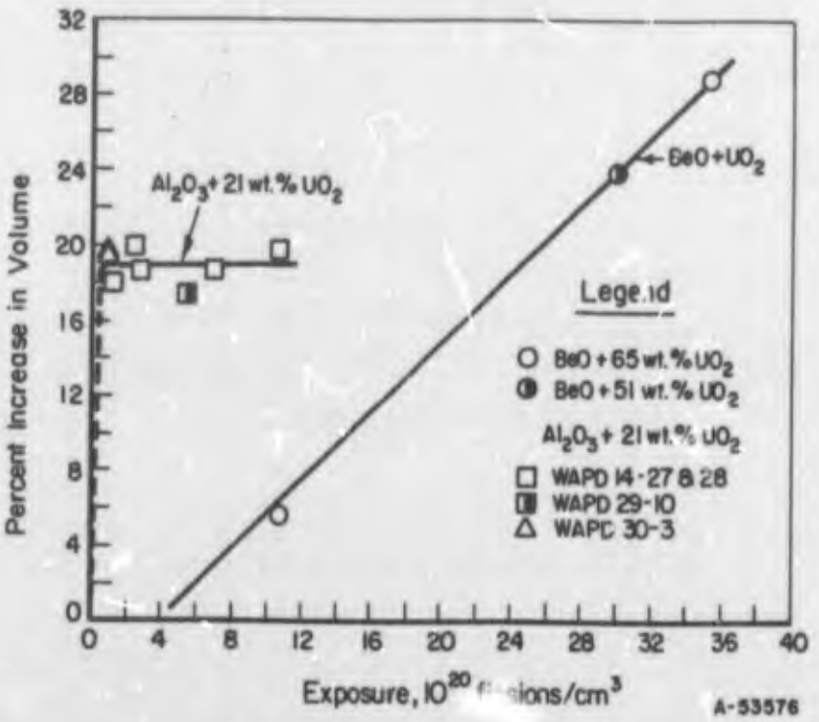


FIGURE 5. SWELLING OF BeO-UO_2 and Al_2O_3-UO_2 AS A FUNCTION OF BURNUP[5]

Comparisons were also made in the performance of dispersions from which the UO_2 at the surface had been leached away by acid. Both of these specimens contained 150-μ-size UO_2 particles and were irradiated at similar conditions. It was found that leached specimens underwent larger dimensional changes than the unleached specimens. No reason was given for this behavior. Metallographic examination of specimens containing coarse UO_2 particles showed that the BeO surrounding the UO_2 particles was damaged. The amount of damage is believed to be limited by the distance that fission fragments travel in BeO (Figure 6). X-ray-diffraction studies that were performed on the irradiated specimens did not show either the BeO or UO_2 crystal structure, indicating that the structure of both may have been destroyed by irradiation.

Considerable amounts of BeO-UO_2 have been irradiated in conjunction with developing the ML-1 (Mobile Gas-Cooled Reactor) Reactor for the United States Army.

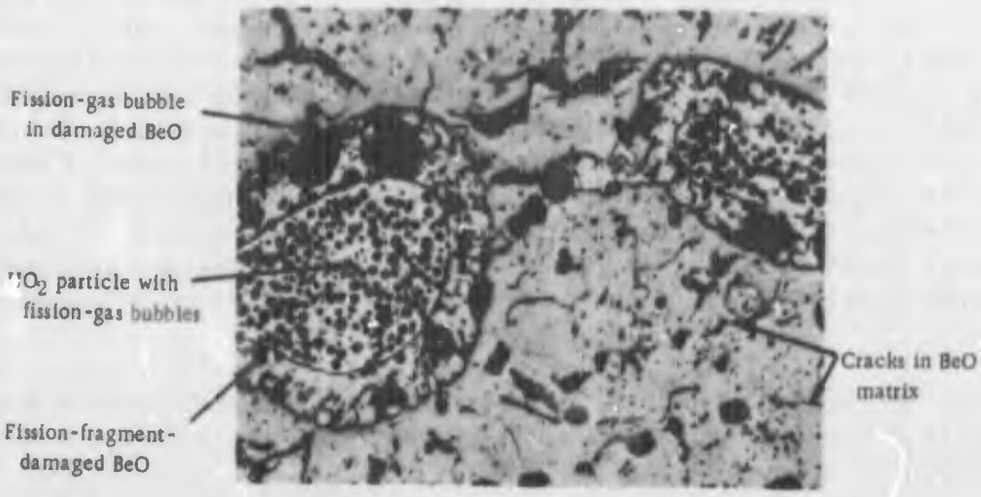


FIGURE 6. FISSION-FRAGMENT DAMAGE IN THE BeO MATRIX AS A RESULT OF IRRADIATION[4]

The specimen received a burnup of 11×10^{20} fissions/cm^3.

Two fuel-element bundles, each of which contained 12 BeO-60 weight percent UO_2 fuel pins, were irradiated by Aerojet-General Nucleonics.[6] These sintered fuel pellets had a diameter of 0.180 inch and were contained in 0.030 inch of Hastelloy X. The structure of the fuel showed a continuous phase of both BeO and UO_2. Irradiation of these pins at surface temperatures of about 955 C to a maximum burnup of 5.7×10^{20} fissions/cm^3 did not result in any specimen failures. However, some bowing of the 28-inch-long specimens, which was attributed to constraint imposed by spacer wires during thermal expansion of the fuel pins, took place. The specimens released less than 1 percent of their fission gases. Metallographic examination of the fuel showed that some cracking hau taken place. Agglomeration of the UO_2 into large particles occurred in some of the specimens.

Capsule BRR-CCR-2, containing six BeO-56 weight percent UO_2 dispersions, was irradiated for 1150 hours at 930 C.[7] No significant changes were observed in specimen dimensions or microstructure of the irradiated fuel. Fission-gas release from the specimens was rather low.

Specimens containing 70 to 80 weight percent UO_2 were irradiated as part of MTR-GCR-3.[8] In these specimens it was not possible to fix a meaningful UO_2 particle size since both the UO_2 and BeO formed continuous phases. Six specimens were irradiated at high temperatures to burnups of up to 6.3×10^{20} fissions/cm^3 of fuel. One of the six specimens failed by rupture after a diametrical increase of 15 percent had occurred. The other specimens underwent minimal dimensional changes. Fission-gas release from the specimens was 0.6 to 2.8 percent, except the specimen that ruptured, which released 69 percent of its fission gases. Metallographic examination of the specimens revealed that central void was formed in one of the specimens, indicating higher than expected central fuel temperatures. Other changes that were

observed were bubble formation and the destruction of UO_2 microstructures near the outer edge of the fuel pellets. This seems to indicate that nonuniform fissioning took place in the fuel, the outer edge receiving a much higher burnup than the center. Because of the high uranium-235 enrichment in the high UO_2 loading, it is quite possible that the center of the fuel underwent a very low burnup, because most of the available neutrons were attenuated by the fuel at the outer edge of the specimens. A damaged zone of BeO was found at the interface of UO_2 and BeO, indicating fission-fragment damage. However, X-ray diffraction studies showed the BeO and UO_2 crystal structure to be essentially undamaged. It must be taken into consideration that fissioning in the samples was not uniform, and consequently the results represent averaged conditions.

Bundle 1B-14R contained 19 BeO-65 weight percent UO_2 fuel pins and was irradiated for 3730 hours.[7] Results of the test indicated that no dimensional or microstructural changes in the fuel occurred.

Further irradiation of another bundle of BeO-UO_2 fuel pins was carried out (1B-17R-2). These specimens contained large UO_2 particles (100 to 500 μ), and the UO_2 content was 75 weight percent. Irradiation of these Hastelloy X-clad pins was carried out for 5820 hours at a maximum surface temperature of 770 C, the maximum burnup being 3.3×10^{20} fissions/cm^3. The specimens did not undergo any dimensional changes, and the fission-gas release was 1 to 2.2 percent. Considerable cracking was found in both the fuel and the matrix. The cuase of this cracking was believed to be due to the thermal-conductivity mismatch of BeO and UO_2.[9]

A capsule containing three BeO-70 weight percent UO_2 specimens among different fuel specimens was irradiated.[10] The irradiation temperature of the specimens varied from 800 to 940 C, the burnup being from 12.4 to 14.8×10^{20} fissions/cm^3 of fuel. Examination of the specimens after irradiation revealed that the cladding on two of the three specimens had cracked. The fuel pellets had undergone diameter increases of 3.3 to 6.4 percent, with the density decrease being 8.4 to 11.1 percent. The unclad specimen released about 13.4 percent of its fission gases. During metallographic examination of the fuel it was found that considerable "pull-out" of the UO_2 particles occurred. This is attributed to fission-fragment damage in the surrounding BeO matrix. X-ray-diffraction studies on the irradiated BeO-70 weight percent UO_2 showed that the UO_2-lattice parameter had increased slightly. Only faint traces of the BeO structure were detected.

General Atomics has investigated BeO-UO_2 as a possible fuel for EBOR (Experimental Beryllium Oxide Reactor) and MGCR (Maritime Gas Cooled Reactor).

A total of eight specimens of BeO-UO_2-60 weight percent were irradiated.[11] Four of these specimens contained 25-μ UO_2 particles, while the other four contained 150-μ particles. The specimens were irradiated for a total of 11,700 hours at maximum center temperatures of 1095 C and burnups of 3.7×10^{20} fissions/cm^3. Cracking of the Monel cladding prevented analysis of fission-gas from seven of the specimens, but one analysis from the fine-particle specimens showed that only 1 percent of the fission gases was released. Examination of the specimens showed that the specimens containing the 150-μ UO_2 particles were intact, while the specimens containing the 25-μ UO_2 particles had cracked. Dimensional increases of the coarse fuel particles were 0.4 to 0.8 percent, corresponding to density decreases of 1.5 to 3 percent. Since the fine-particle specimens were cracked, no dimensional measurements on them were possible. Metallographic examination of the specimens showed that no gross microstructural changes had occurred as a result of irradiation.

As part of the Maritime Gas Cooled Reactor program, three fuel assemblies were irradiated in the D-R gas loop located in a Hanford Reactor.[12] The fuel in these assemblies consisted of BeO-60 weight percent UO_2, with the particle size varying from 1 to 150 microns. Tests were conducted at 820 and 930 C for times up to 1430 hours. It was found that the fission-gas release from the fuel was minimal, and that no dimensional changes took place.

Six 0.160-inch-diameter BeO-56 weight percent UO_2 specimens containing 40-μ UO_2 particles and clad with 30 mils of Hastelloy X, were irradiated at surface temperatures of 790 to 930 C.[13] The specimens underwent burnups of 0.83 to 1.4×10^{20} fissions/cm^3 of fuel. The clad specimens underwent no dimensional changes as a result of irradiation. Two of the six specimens were sectioned, and the fuel pellets were removed. Density measurements performed on the pellets showed that the individual pellets had undergone density decreases of 1 to 2.4 percent. No microstructural changes were caused in the fuel by irradiation.

A total of 28 pellets encapsulated in four different capsules were irradiated.[14-16] The sintered specimens contained 47 to 54 weight percent UO_2 and had densities of 89 to 95 percent of theoretical density. Average UO_2 particle size was about 125 microns. All of the specimens were irradiated at surface temperatures of 815 C, the average fuel temperature being 955 to 1025 C. Length of irradiation on the specimens varied from 1150 to 10,300 hours, with the maximum burnup being 5.8×10^{20} fissions/cm^3. The specimens were found to undergo no dimensional changes at burnups of 9×10^{19} fissions/cm^3, but did undergo progressively larger dimensional and density changes with higher burnup and longer irradiation time, the maximum being a 5.5 percent density decrease after 5×10^{20} fissions/cm^3. The crushing strength of the pellets was reduced from 86,300 psi to 68,500 psi by an irradiation of 2×10^{20} fissions/cm^3. Metallographic examination of the fuel irradiated to 2×10^{20} fissions/cm^3 did not show any major microstructural changes except for an appearance of damaged BeO next to UO_2 particles. It was also found by X-ray-diffraction studies that the BeO crystal structure was not destroyed by the irradiation. Resistance to abrasion when tested by grinding was also reduced as a result of irradiation.

An irradiation experiment containing BeO-60 weight percent UO_2 specimens was performed to study the effect of UO_2 particle size on irradiation performance, especially on fission gas release.[14,17] Three of the sintered specimens contained 25-μ UO_2 particles, while the other three contained 150-μ UO_2 particles. Both types of specimens were irradiated to burnups of 2 to 2.3×10^{20} fissions/cm^3 at surface temperatures of 930 C. The specimens underwent density decreases of 2.4 to 4.5 percent; both types of specimens performed about equally. No major changes were observed in the irradiated microstructure, although the BeO that surrounded the UO_2 particles tended to "pull-out" during the metallographic preparation. The specimens that contained the 150-μ UO_2 particles were found to release 17 to 24 percent of their fission gases, while the 25-μ UO_2 particles released only 0.3 percent. These results indicate that diffusion of fission gases in UO_2 is probably faster than diffusion of fission gases in BeO. This is based on the reasoning that the fission gases from smaller particles are recoiled into the BeO matrix, from which they would have to be released by diffusing through the BeO. In the case of large UO_2 particles, the fission fragment

comes to rest in the UO_2 particle itself and consequently diffuses through the UO_2 matrix to be released. It is thought that in both cases the interface between the UO_2 particle and BeO matrix acts as a sink for fission gases. These interfaces are interconnected to the outside, and therefore any fission-gas atom reaching the interface can be considered as released to the outside. This finding is supported by another investigation where specimens containing 10-, 150-, and 400-μ UO_2 particles dispersed in BeO were annealed at 955, 1180, and 1370 C after irradiation.[18] The fission-gas release during annealing was measured, and the results indicated that the fission-gas release increased with increased particle size. Attempts were made to improve the fission-gas retention properties of BeO-80 weight percent UO_2 compacts by precoating 150-μ UO_2 particles with BeO before sintering.[19,20] However, it was found that these compacts resulted in low density (94 percent of theoretical), and the fission-gas release was about 0.3 to 2 percent at 1600 C, which is comparable to BeO-80 weight percent UO_2 that did not contain precoated specimens. The difference between the structure with precoated UO_2 particles and the usual structure resulting from sintering is shown in Figure 7.

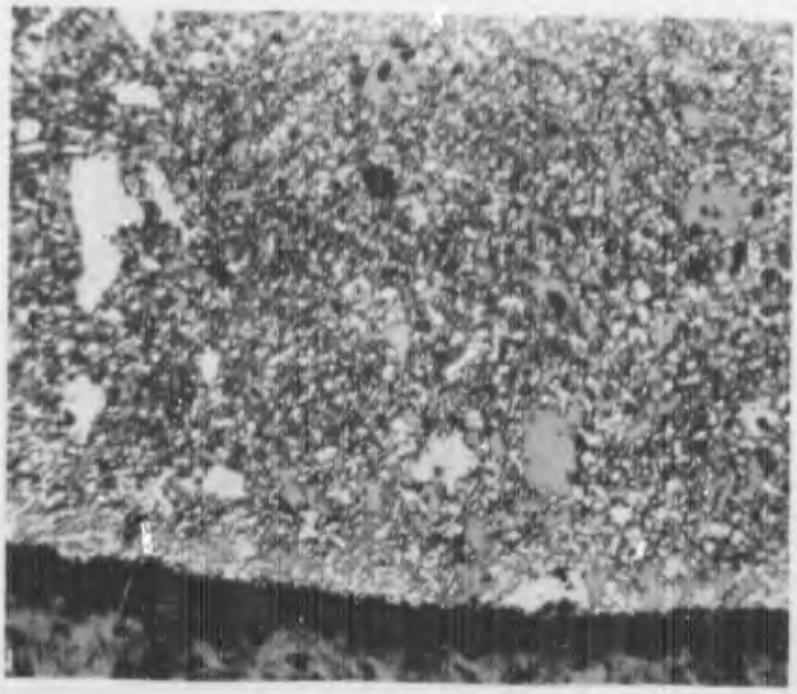
At Oak Ridge, sintered tubular specimens of BeO-50 weight percent UO_2 were irradiated.[21] The specimens contained rather fine UO_2 particles (0.5 to 5 μ) in the BeO matrix. The center fuel temperature during irradiation was 1200 C and the maximum burnup was 3.7×10^{20} fissions/cm^3. The specimens released only 0.2 to 0.3 percent of their produced fission gases but underwent considerable dimensional changes. The maximum dimensional changes were 2 and 2.5 percent increases in diameter and length, respectively. The maximum density decrease was 8.7 percent. It was not possible to distinguish the UO_2 particles after irradiation, this being caused by the rather fine particle size of UO_2 and porosity in the fuel.

Another capsule containing BeO-62 weight percent UO_2 with UO_2 particles measuring about 150 μ, was also irradiated.[22,23] These specimens were irradiated at surface temperatures of 370 C and fuel center temperatures of 1160 C to a burnup of 1.1×10^{20} fissions/cm^3. Examination of the irradiated specimens showed a density decrease of about 1 percent. Metallographic examination revealed that the average UO_2 particle size had increased considerably in the center of the specimen.

Irradiations of BeO-UO_2-ThO_2 have been carried out at Oak Ridge National Laboratory and Lucas Heights, Australia.

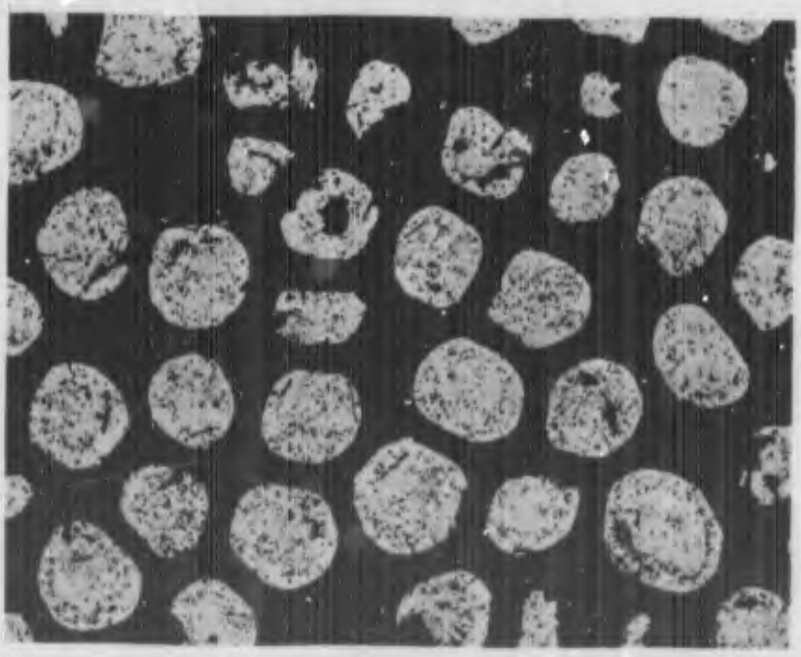
At Oak Ridge, specimens containing BeO-35.5 weight percent ThO_2-25.4 weight percent UO_2 were irradiated, together with BeO-UO_2 specimens.[21] The specimens contained 0.5 to 5.0 μ of ThO_2-UO_2 and were sintered to a density of 96 percent of theoretical density. Irradiations took place at average fuel temperatures of 1095 C, and the maximum burnup was 3.1×10^{20} fissions/cm^3. The dimensional changes and fission-gas release of the BeO-ThO_2-UO_2 specimens were considerably less than those experienced by BeO-UO_2 specimens at the same irradiation temperatures and burnups. Irradiation performances of BeO-ThO_2-UO_2 dispersions are given in Table 2.

The specimens irradiated by the Australians contained varying amounts of UO_2-ThO_2 solid solution dispersed in BeO.[24] The UO_2 content ranged from 2.6 to 15.2 percent, while the ThO_2 content varied from 2.8 to 39.4 percent. Hot-pressed and



250X HC12201

a. Dispersion With a Continuous UO_2 Phase
(98% Theoretical Density)



100X RM28951

b. Dispersion With Precoated UO_2 Particles
(94% Theoretical Density)

FIGURE 7. COMPARISON OF DIFFERENT BeO-80 WEIGHT PERCENT UO_2 MICROSTRUCTURES[19, 20]

TABLE 2. IRRADIATION PERFORMANCE OF BeO-$(Th,U)O_2$ FUEL RODS[a,b]

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume %	UO_2, weight per-cent	ThO_2, weight per-cent	Density, % T.D.[c]	Fuel Particle Size, μ	Cladding Material	Average Temp, C	Burnup, 10^{19} fission/cm^3	Bulk Heat Gen. Rate, kw/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %				Reference
													Diameter	Density	Volume	Length	
100/101	2	9.4	3	2.6	6.8	94-99	100-185	Sodium	570-670	2.0	0.065	0.25	0.25		0.7	0.7	24
102/105	2	9.4	3	2.6	6.8	94-99	100-185	Sodium	570-670	2.0	0.054	0.28	0.2		0.6	0.15	24
106/107	2	13.4	4.4	3.7	9.7	94-99	100-185	Sodium	600-650	3.5	0.099	0.1	0.15		0.35	0.07	24
108/111	2	13.4	4.4	3.7	9.7	94-99	100-185	Sodium	600-650	3.5	0.076	0.1	0.15		0.4	0.1	24
112/113	2	24.1	8.4	6.7	17.4	94-99	100-185	Sodium	640-700	6.5	0.194	0.1	0.1		0.8	0.7	24
114/117	2	24.1	8.4	6.7	17.4	94-99	100-185	Sodium	640-700	6.5	0.158	0.12	0.2		0.6	0.2	24
118/119	2	39.2	15.7	10.9	28.3	94-99	100-185	Sodium	600-650	10	0.300	0.37	0.15		0.65	0.35	24
120/123	2	39.2	15.7	10.9	28.3	94-99	100-185	Sodium	600-650	10	0.256	–	0.05		0.4	0.3	24
124/125	2	54.6	25.8	15.2	39.4	94-99	100-185	Sodium	720-835	15	0.415	0.26	0.15		0.5	0.2	24
126/129	2	54.6	25.8	15.2	39.4	94-99	100-185	Sodium	720-835	15	0.345	0.16	0.07		0.3	0.15	24
130/131	2	15.1	4.4	7.4	7.7	94-99	100-185	Sodium	670-730	6	0.168	–	0.15		0.45	0.15	24
132/135	2	15.1	4.4	7.4	7.7	94-99	100-185	Sodium	670-730	6	0.148	0.1	0.3		0.8	0.2	24
139/140	2	14.7	4.4	11.9	2.8	94-99	100-185	Sodium	640-690	8	0.225	0.25	0.25		0.75	0.25	24
141/147	2	14.7	4.4	11.9	2.8	94-99	100-185	Sodium	640-690	8	0.205	0.23	0.25		0.65	0.15	26
148/149	2	13.8	4.4	3.7	10.1	94-99	<10	Sodium	580-630	3	0.089	0.1	0.35		0.85	0.15	26
150/153	2	13.8	4.4	3.7	10.1	94-99	<10	Sodium	580-630	3	0.080	0.1	0.45		1.1	0.2	26
154	1	15.2	5	4.1	11.1	94-99	100-200	Sodium	610-725	3		0.1	0.1	-0.6	0.35	0.15	26
155	1	15.2	5	4.1	11.1	94-99	100-200	Sodium	610-725	3.3		0.1	0.2	-0.65	0.6	0.2	26
158	1	37.7	15	11.1	27.6	94-99	100-200	Sodium	670-795	8.9		0.6	0.15	-0.65	0.45	0.15	26
159	1	37.7	15	11.1	27.6	94-99	100-200	Sodium	670-795	8.1		1.1	0.2	-0.75	0.65	0.25	26
160	1	37.7	15	11.1	27.6	94-99	100-200	Sodium	675-780	9.0		0.65	0.15	-0.55	0.45	0.15	26
161	1	37.7	15	11.1	27.6	94-99	100-200	Sodium	675-780	9.1		1.4	0.2	-0.75	0.6	0.2	26
264	1	38.3	15	20	18.3	94-99	100-200	Sodium	720-850	14		1.8	0.2	-0.65	0.6	0.2	26
265	1	38.3	15	20	18.3	94-99	100-200	Sodium	720-850	14		2.6	0.2	-0.6	0.7	0.3	26
272	1	38.6	15	29.6	19	94-99	100-200	Sodium	725-820	18		1.4	0.3	-1.1	0.9	0.3	26
273	1	38.6	15	29.6	19	94-99	100-200	Sodium	725-820	20		0.9	0.35	-1.2	1.0	0.3	26
279	1	59.5	30	43.6	15.9	94-99	100-200	Sodium	730-875	14		1.8	0.2	-0.55	0.6	0.2	26
280	1	59.5	30	43.6	15.9	94-99	100-200	Sodium	730-875	14		1.5	0.25	-0.7	0.75	0.25	26
283	1	38.6	15	29.6	19	94-99	<20	Sodium	720-940	17		0.1	0.25	-1.2	(broken)		26
284	1	38.6	15	29.6	19	94-99	<20	Sodium	720-940	18		0.1	0.15	-1.4			
LM-3a	1	60	29	25.4	35.5	96	0.5-5	Inconel	930	31		0.3	0.3	-2.1		0.2	21
LM-3b	1	60	29	25.4	35.5	96	0.5-5	Inconel	1095	31		0.01	1.5			2.2	21
LM-5a	1	60	29	25.4	35.5	96	0.5-5	Inconel	815	19		0.08	0.3			0.3	21
LM-5b	1	60	29	25.4	35.5	96	0.5-5	Inconel	1040	19		0.1	0.6			0.2	21

(a) Length of specimens in References 24 and 26 is either 0.59 or 1.18 in.
(b) Diameter of specimens in References 24 and 26 is either 0.296 or 0.354 in.
(c) T.D. indicates theoretical density.

sintered specimens were made with the ThO_2-UO_2 solid-solution particles being 100 to 185 μ in some specimens, and less than 10 μ in others. These pellets were irradiated at temperatures varying from 570 to 835 C to a maximum burnup of 1.5×10^{20} fissions/cm^3 of fuel. As a result of irradiation, the specimens underwent volume increases of less than 1 percent and the fission-gas release from the specimens was only 0.1 percent. Although not enough comparative tests were made between the specimens containing fine and coarse fuel particles, the limited number of fine-particle specimens that were tested showed more volume increase as a result of irradiation. Both types of specimens showed about the same fission-gas release. X-ray-diffraction studies on the irradiated specimens showed that the BeO structure was still present, but that line broadening had taken place.[25] Annealing at 1000 C removed the line broadening from the BeO that contained the coarse fuel particles, while annealing at 1350 C was necessary to remove the line broadening from the BeO containing the fine particles.

Cold-pressed and sintered dispersions containing ThO_2-UO_2 particles of 100 to 200 μ and >20 μ in BeO were irradiated at temperatures of 610 to 875 C to a maximum burnup of 2×10^{20} fissions/cm^3.[26] The specimens underwent minor dimensional and density changes; those specimens that received the highest fission density underwent somewhat larger changes. Fission-gas release ranged from 0.1 to 4.8 percent for the specimens containing coarse fuel particles, with no correlation to temperature and fission density; however, specimens with fine fuel particles released less than 0.1 percent of the fission gases. The only microstructural change appeared to be "pull-out" of the coarse fuel particles. This is attributed to the damage of the surrounding BeO matrix by fission fragments. Annealing of the coarse-particle fuel at 1250 C revealed the formation of fine pores at the grain boundaries. Annealing at 1500 C caused considerable coalescence of the bubbles in both the grain boundaries and also in the grains. Grain pullout was eliminated by annealing at 1500 C. No microstructural changes were detected in the fine-particle fuel after annealing at 1250 and 1500 C. This was attributed to the fine particle and grain size.

Aluminum Oxide

Fuel plates containing Al_2O_3-UO_2 dispersions have been irradiated at Westinghouse.[3] These dispersions contained 5-μ UO_2 fuel particles and were irradiated at a center temperature of 780 C to burnups up to 5.2×10^{20} fissions/cm^3. The fuel plates underwent a 20 percent volume increase at 9.2×10^{19} fissions/cm^3 of fuel, but no further volume increase took place when irradiated to 5.2×10^{20} fissions/cm^3 of fuel (Figure 5). Irradiation performances of Al_2O_3-UO_2 dispersions are given in Table 3. Metallography of the irradiated specimens revealed the loss of grain boundaries. The Al_2O_3 crystal structure could not be detected by X-ray studies. Fission-gas release from the dispersion was found to be only 1 percent.

Three specimens containing Al_2O_3-20 weight percent UO_2 were irradiated by General Atomics at 815 C to a burnup of 0.45×10^{20} fissions/cm^3.[27] Examination of the specimens showed that the diametral increases of 0.26 to 0.78 percent had occurred as a result of irradiation.

TABLE 3. IRRADIATION PERFORMANCE OF Al_2O_3-UO_2 FUEL ELEMENTS

Sample Designation	Number of Specimens in Sample	Fuel Content, %	Fuel Volume, %	Density, % T.D.[a]	Shape	Length, in.	Diam. or Width and Thickness, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature, C		Irrad. Time, hr	Burnup 10^{20} fissions/cm^3 of fuel	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %		Reference
											Surface	Center						Diam.	Volume	
29-10-1	(4)	21	4	96	Plate	1.0	0.51 x 0.038	5	Zr-2	.017	230	740	2640	5.2	4.9	195	1.0		17.4	5
30-3-T04	(42)	21	4	96	Plate	0.61	0.61 x 0.06	5	Zr-2	.035	280	780	870	0.9	2.0	160			19.5	5
T 17	(42)	21	4	96	Plate	0.61	0.61 x 0.06	5	Zr-2	.025	280	780	870	0.9	2.0	160			19.5	5
1	1	44	20		Pin		0.247		Ni	.030	815		2000	0.45	0.66	50		0.78		27
2	1	44	20		Pin		0.247		Ni	.030	815		2000	0.45	0.66	50		0.11		27
3	1	44	20		Pin		0.247		Ni	.030	815		2000	0.45	0.66	50		0.26		27

(a) T.D. indicates theoretical density.

Postirradiation annealing studies were performed on four specimens containing 20 volume percent UO_2 particles dispersed in Al_2O_3.[28] These specimens were irradiated for 1,520 hours at 815 C before annealing for 20 hours at 980, 1260, and 1430 C; annealing resulted in fission-gas releases of 0.8, 12, and 45 percent, respectively.

Magnesium Oxide

The British have found that MgO-UO_2 dispersions have rather high in-pile fission-gas release.[29] Magnesium oxide containing 0.1-μ UO_2 particles was found to release 2 percent of its fission gases at 600 C, with the release increasing to nearly 100 percent at 800 C. When 100-μ UO_2 particles were used, the fission-gas release was about 25 percent at 800 C. The rather high fission-gas releases were probably due to the low density (80 percent of theoretical density) of the dispersions.

Graphite

Graphite has been used as a matrix material because of its high strength at elevated temperatures, high melting point, high sublimation temperature, excellent resistance to thermal stresses, and its good moderating ability. The main drawback to using graphite appears to be its poor resistance to oxidizing atmospheres and carburization of cladding materials at high operating temperatures. There are two distinctly different methods of fabricating fueled graphite specimens.

To prepare the specimens by "impregnation", porous graphite is impregnated by a solution containing uranyl nitrate or ammonium diuranate. After a sufficient soaking time the composite is heated at about 800 C to form UO_2. The main difficulty with this process is that the fuel particles are of small size, being only about 1 μ.

The other method of fabrication is to sinter graphite powder with UO_2 powder of the desired particle size. An organic binder is usually included to set the particles, after which they are pressed at 20,000 psi and heated at about 1400 C.

Dispersions of UC and UC_2 can be placed directly into the graphite matrix by sintering, or UO_2 particles can be carburized to uranium carbides at sufficiently high temperatures. However, if the particle size is about 100 μ, enough carbon is removed from the immediate area to weaken the composite and cause crumbling of the specimen.

A study was undertaken at Oak Ridge to determine the effect of the fuel-particle size on irradiation performance of dispersed fuels in graphite.[2] Uranium dioxide particles of 16-, 20-, 94-, 334-, and 586-μ diameter were dispersed in graphite and irradiated at about 65 to 85 C for 550 to 610 hours. Burnups of the fuel particles were about 0.1 atom percent of uranium. After irradiation, the weight, modulus of elasticity, electrical resistivity, and thermal conductivity were measured and compared to these properties before irradiation. It was found that the smallest sized particles had the largest effect on these properties. This is illustrated in Figure 8, which shows the effect of irradiation on electrical resistivity of graphite-UO_2 dispersions containing

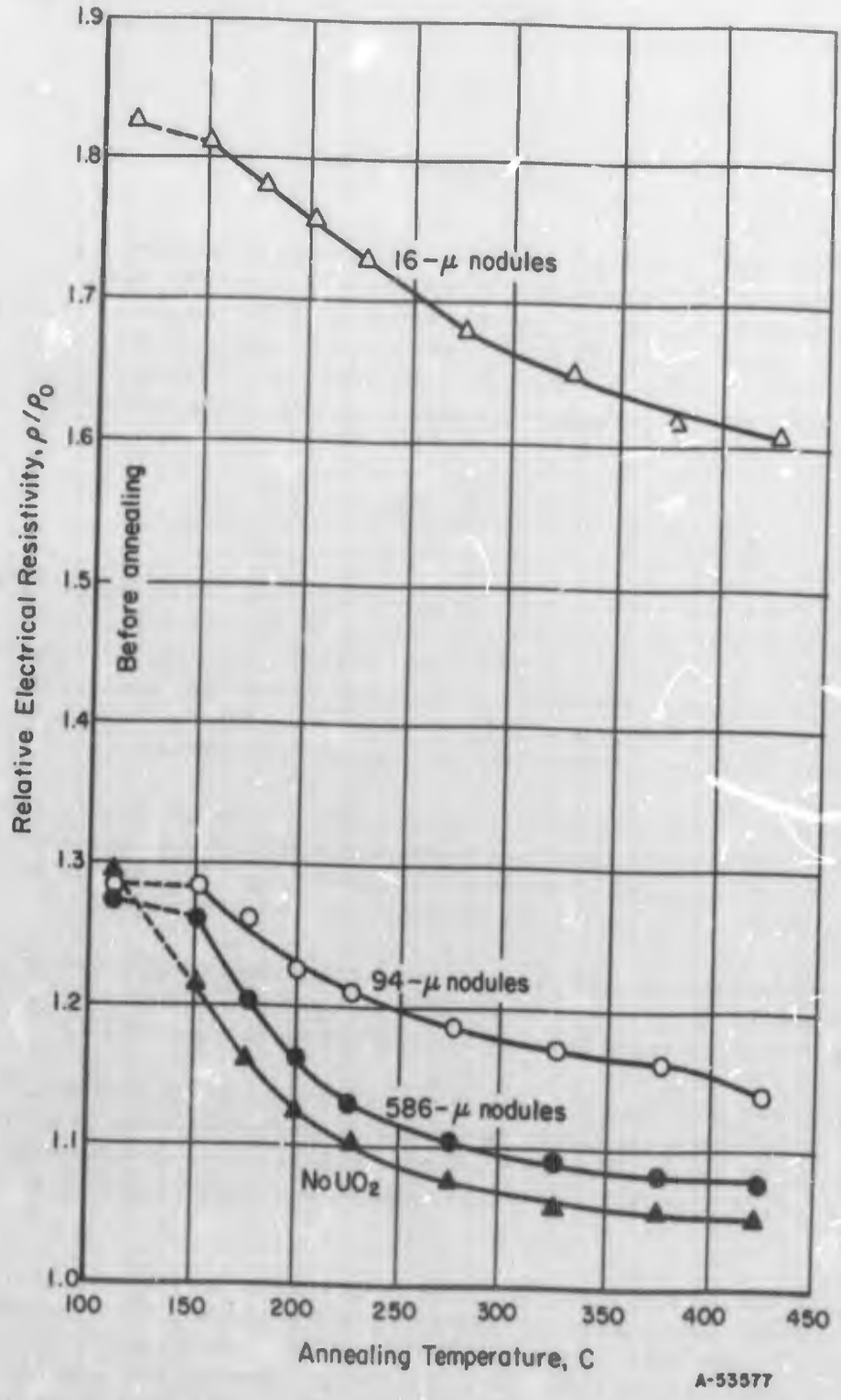


FIGURE 8. ANNEALING OF RADIATION DAMAGE FROM GRAPHITE-UO_2 DISPERSIONS[2]

various sizes of UO_2 particles. Annealing studies showed that the damage caused by fast neutrons (specimens with large UO_2 particles) was annealed out at 300 C, while the damage caused by fission fragments (specimens with small UO_2 particles) was not annealed out at 750 C.

Other studies have shown that thermal conductivity of fueled graphite is considerably reduced by irradiation (Figure 9).[30,31] It has been found that the smaller the particle size the larger the reduction in thermal conductivity at the same irradiation temperature and burnup. Irradiation temperature is important because at temperatures above 1000 C most of the irradiation damage, which contributed to a decrease in thermal conductivity, is annealed out.

Fueled 1.5-inch-diameter graphite spheres were irradiated to determine stability of dispersed fuel for the proposed Pebble Bed Reactor (PBR).[32] Both "impregnated" and sintered fuel specimens were irradiated. These specimens were irradiated at surface temperatures of 650 to 730 C to burnups of to 4×10^{18} fissions/cm^3 for the "impregnated" specimens, and 2×10^{20} fissions/cm^3 for the sintered specimens. As a result of the irradiation, the specimens underwent diametral decreases of 0.8 to 1.0 percent for the "impregnated" specimens and 0.07 to 0.37 percent for the sintered specimens. This shows that the sintered specimens are considerably more stable, since they underwent smaller dimensional changes at the same temperatures and higher burnups. Mechanical tests performed on the specimens indicate an increase in compressive strength and a decrease in impact strength for both types of specimens.

Four fuel elements containing UC_2 dispersed in graphite were irradiated.[33,34] The UC_2 particle size varied from 160 to 240 μ, and the UC_2 content in one dispersion was 10.5 weight percent and 19.3 weight percent in the other. The specimen surface temperatures during irradiation were 710 to 865 C, and the maximum burnup was 7.7×10^{19} fissions/cm^3 of fuel. Diameter decreases were from 4.2 to 5.1 percent, while the length increases were 3.0 to 5.2 percent. Density of the specimens increased from 2.5 to 5.1 percent. This density increase is probably due to densification of graphite on irradiation at temperatures above 400 C.[35,36] Irradiation performances of graphite-fueled dispersions are given in Table 4. No microstructural changes were noted in the specimens as a result of irradiation.

As part of the Maritime Gas Cooled Reactor (MGCR) program, specimens containing UC and UC_2 dispersed in graphite were irradiated.[27,37] These dispersion specimens contained 61 to 66 weight percent of uranium carbide particles, and were irradiated at surface temperatures of 670 to 830 C. Burnups of the specimens were about 8.7×10^{19} fissions/cm^3 of fuel. During examination of the fuel specimens, it was found that they underwent diametrical decreases of 1.2 to 2.0 percent. Fission-gas release from the specimens was less than 0.5 percent. No apparent changes in the microstructure were caused by the irradiation.

A considerable number of fuel particles dispersed in graphite have been irradiated by General Atomics as part of their development work for High-Temperature Gas Cooled Reactor (HTGR).[38]

In their first test series (GA-308-2), 11.4 weight percent of $(Th,U)C_2$ was dispersed in graphite as 110- to 250-μ fuel particles. The specimens had a diameter of 1.0 inch and were 1.1 inches long. These specimens were irradiated at surface temperatures of 775 to 830 C, the central fuel temperature being an estimated 275 C higher. Burnup of the specimens showed that the specimens were in excellent condition and had undergone dimensional changes of less than 0.2 percent.

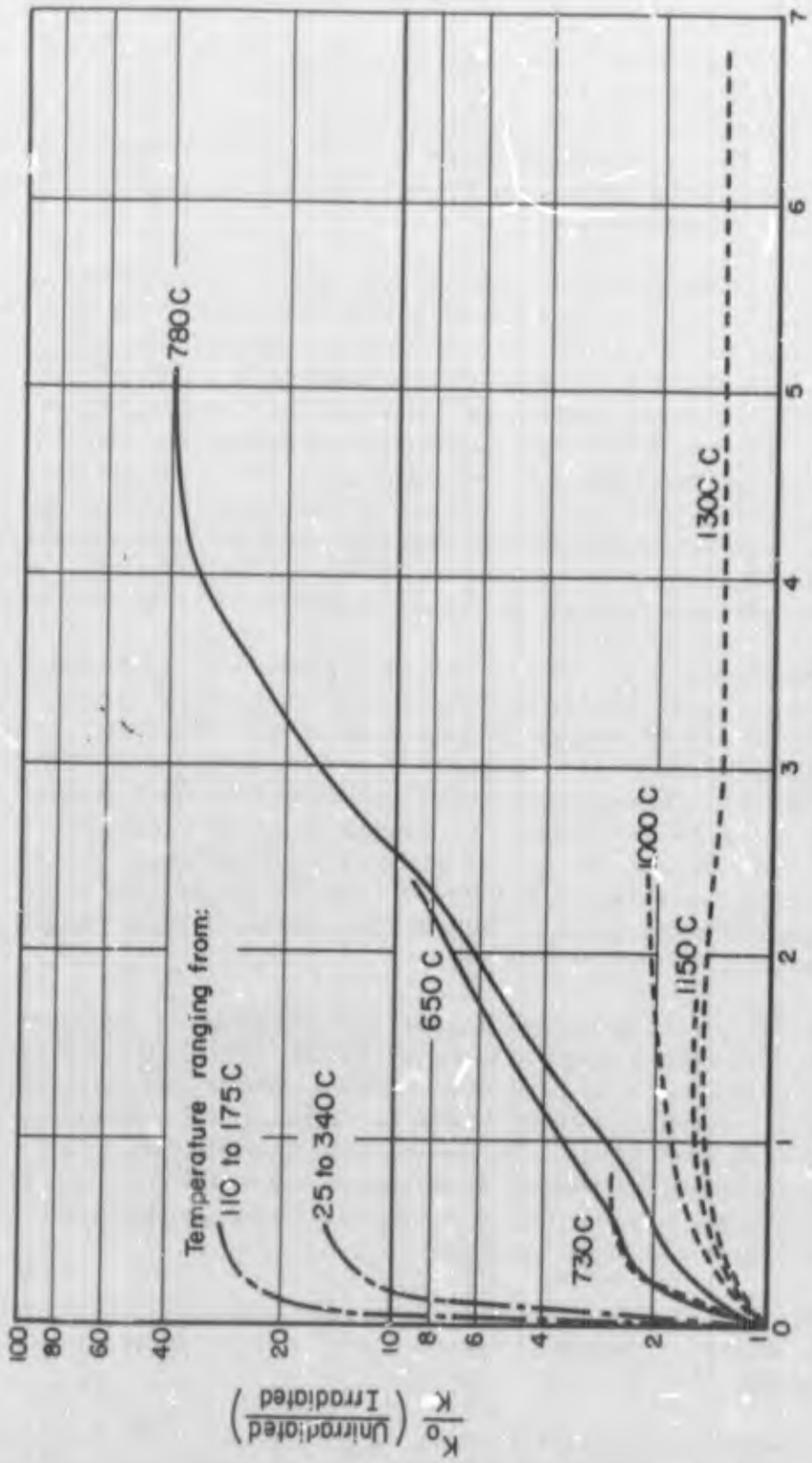


FIGURE 9. IN-PILE THERMAL CONDUCTIVITY OF GRAPHITE-UC DISPERSIONS AT VARIOUS TEMPERATURES AND BURNUPS(30)

TABLE 4. IRRADIATION PERFORMANCE OF VARIOUS NUCLEAR FUELS DISPERSED IN A GRAPHITE MATRIX

Sample Designation	Number of Specimens in Sample	Fuel Material	Fuel Weight, %	Fuel Volume, %[a]	Specimen Shape	Fuel Particle Size, μ	Diameter, in.	Length, in.	Cladding Material	Cladding Thickness, in.	Temperature, C Surface	Temperature, C Center	Irrad. Time, hr	Burnup, 10^{19} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, % Diameter	Density	Volume	Length	Reference
MTR-48-1																						
A	1	UC_2	10.5	1.9	Rod	160-240	1.0	1.5	Graphite		865	1845	1440	6.2	0.27	170		-4.2	2.5		5.2	33
B	1	UC_2	19.3	3.8	Rod	160-240	1.0	1.5	Graphite		710	1990	1440	5.9	0.30	190		-5.2	5.1		3.5	33
MTR-48-2																						
A	1	UC_2	5.6	1	Rod	160-240	1.0	1.5	Graphite		815	1370		5.5	0.14	85		-4.5			3.0	34
B	1	UC_2	8.0	1.4	Rod	160-240	1.0	1.5	Graphite		815	1370		7.7	0.14	89						34
GA-308-2																						
	(6)	$(U,Th)C_2$	11.4[b]	2.5	Rod	110-250	1.0	1.1	Graphite		775-830	1050-1100		4.3								38
GA-308-3																						
	1	$(U,Th)C_2$	24.2	6	Tube	250-500	(c)	1.75	Graphite		650-760	1200		6.9				0.0			0.2	38
	1	$(U,Th)C_2$	24.2	6	Tube	110-250	(c)	1.75	Graphite		730-840	1300		6.9				-0.2			0.2	38
	1	$(U,Th)C_2$	24.2	6	Tube	<50	(c)	1.75	Graphite		840-900	1400		6.9				-1.7			2.2	38
	1	$(U,Th)C_2$	24.2	6	Tube	<50	(c)	1.75	Graphite		900-925	1480		6.9				-2.2			3.0	38
	1	$(U,Th)C_2$	24.2	6	Tube	110-250	(c)	1.75	Graphite		760-840	1200		6.9				-0.3			0.0	38
	1	$(U,Th)C_2$	24.2	6	Tube	250-500	(c)	1.75	Graphite		700-790	1150		6.9				0.0			0.1	38
GA-308-4																						
	1	$(U,Th)C_2$	22.5	5.5		<50					650-760	1090		6.0				-0.9			3.0	38
	1	$(U,Th)C_2$	22.5	5.5		250-500					700-[illegible]	1290		6.0				0.0			0.3	38
	1	$(U,Th)C_2$	22.5	5.5		<50					790-870	1370		6.0				-0.8			0.8	38
	1	$(U,Th)C_2$	22.5	5.5		110-250					870-925	1480		6.0				-0.1			0.1	38
	1	$(U,Th)C_2$	22.5	5.5		250-500					790-840	1230		6.0				0.0			0.1	38
	1	$(U,Th)C_2$	22.5	5.5		110-250					730-815	1175		6.0				0.0			0.1	38
BMI-31-3																						
	1	UC_2	60.0	20.0	Rod		0.22	1.0	316 SS	0.015	690			5.5	0.30	41	0.30	-2.0				27
	1	UC_2	60.0	20.0	Rod		0.22	1.0	316 SS	0.015	830			8.7	0.46	64						27
	1	UC	60.0	20.0	Rod		0.22	1.0	316 SS	0.015	775			6.7	0.35	49		-1.5				27
	1	UC	60.0	20.0	Rod		0.22	1.0	316 SS	0.015	850			8.7	0.45	62	0.37	-1.5		-3.4		27

(a) A graphite density of 1.9 g/cc was used in the calculations.
(b) ThC_2 content is 70 percent of fuel.
(c) Specimens had an outside diameter of 1.0 in. and an inside diameter of 0.5 in.

Capsule GA-308-3 contained specimens which were annular rather than cylindrical. Also the fuel content was increased to 24.2 weight percent. In an effort to determine the effects of fuel particle size on fuel performance, specimens of three different particle sizes were used. These particle sizes were less than 50 μ, 110 to 250 μ, and 250 to 500 μ. During irradiation, the surface temperatures of the specimens varied from 680 to 925 C and the fuel burnup was 6.9×10^{19} fissions/cm^3. Appearance of the fuel bodies after irradiation was good. Dimensional measurements on the irradiated specimens indicated that specimens containing the 250- to 500-μ fuel particles were the most stable, while specimens containing fuel particles less than 50 μ were the least stable. Specimens containing the 110- to 250-μ particles underwent minimal dimensional changes; thus their performance is almost identical to those containing the 250- to 500-μ particles.

In Capsule GA-308-4, the specimens were identical to those of GA-308-3, except that the fuel material was prepared differently. Irradiation temperatures in this capsule varied from 650 to 925 C, and the burnup was 6×10^{19} fissions/cm^3. The appearance of the specimens was excellent except for surface roughening, which took place in a specimen containing 50-μ fuel particles. Again the specimens containing large fuel particles showed better dimensional stability.

Considerable amounts of testing have been carried out to determine the fission-gas-release properties of fueled graphite. Postirradiation-annealing tests on "impregnated" graphite-fueled specimens has shown that considerable amounts of the fission gases are released at the temperatures of interest (1000 to 1800 C).[39] Also, the magnitude of fission-gas release is very much dependent on the fuel-particle size and porosity of the graphite. The smaller the particle size, the larger the fraction of fission products that is recoiled into the graphite matrix, and the more porous the graphite matrix the faster the release of fission gases from the matrix. Figure 10 illustrates the dependence of fission-gas release on temperature and specimen characteristics.

Since the sintered specimens have a larger fuel-particle size (150 μ vs ~1 μ), their fission-gas-retention properties are better than those of the "impregnated" specimens.[40] Figure 11 shows that the fission-gas release increases slightly with the half-life of the radioactive isotope, indicating that the release is diffusion dependent.

Postirradiation-annealing studies were performed on sintered specimens to determine their ability to retain fission gases at various temperatures, and also to determine the effect of particle coating on fission-gas release.[38] Results of these tests are shown in Figure 12. It can be seen that the fission-gas release is very much temperature dependent, and coating of fuel particles improves the ability of the fuels to retain their fission gases.

Therefore, the testing of uncoated fuel particles dispersed in graphite has been discontinued, and the main emphasis of research was shifted to development of coated fuel particles. The performance of coated fuel particles will be discussed in another REIC report.

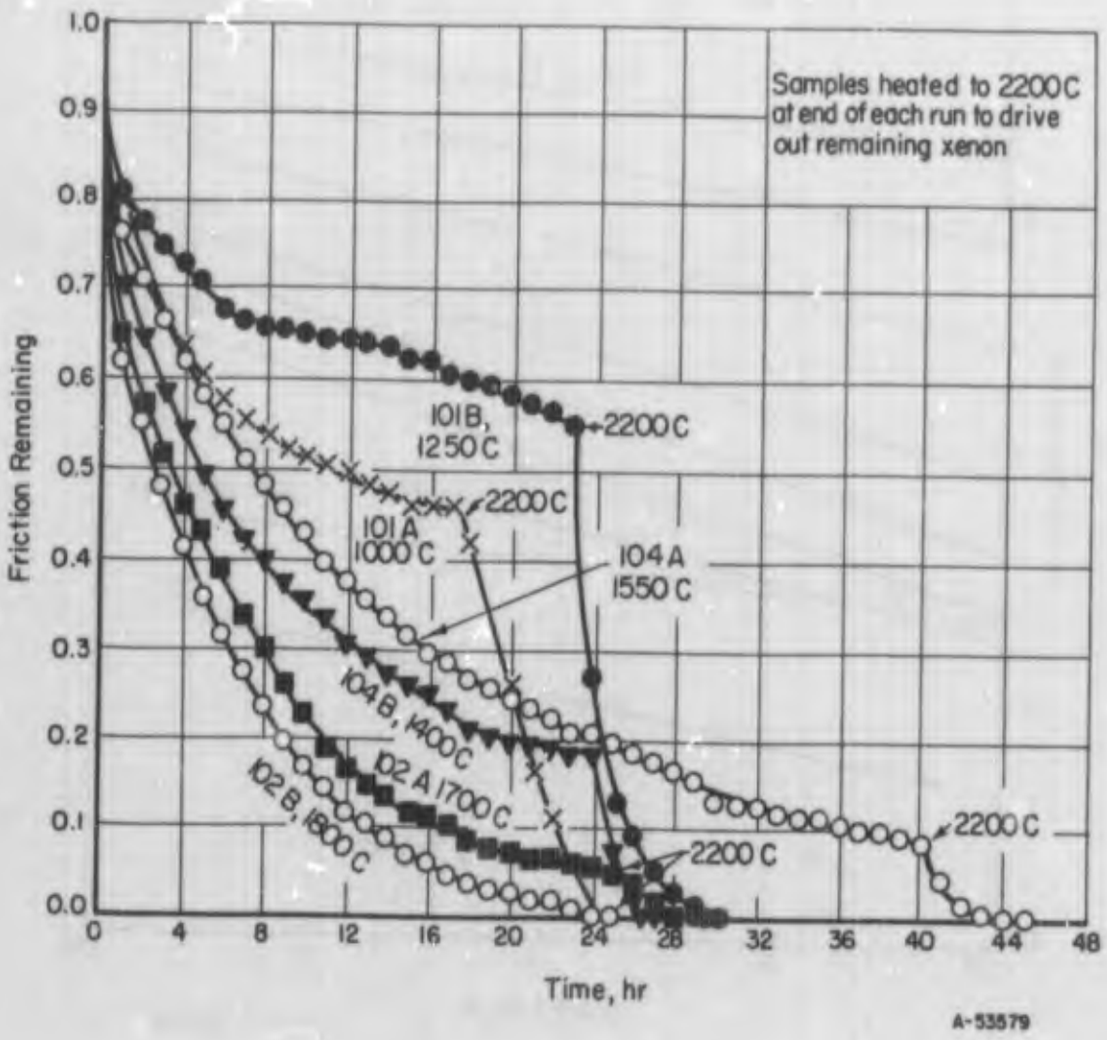


FIGURE 10. Xe^{133} RELEASE FROM TURRET-TYPE UO_2-IMPREGNATED GRAPHITE COMPACTS AT VARIOUS TEMPERATURES[39]

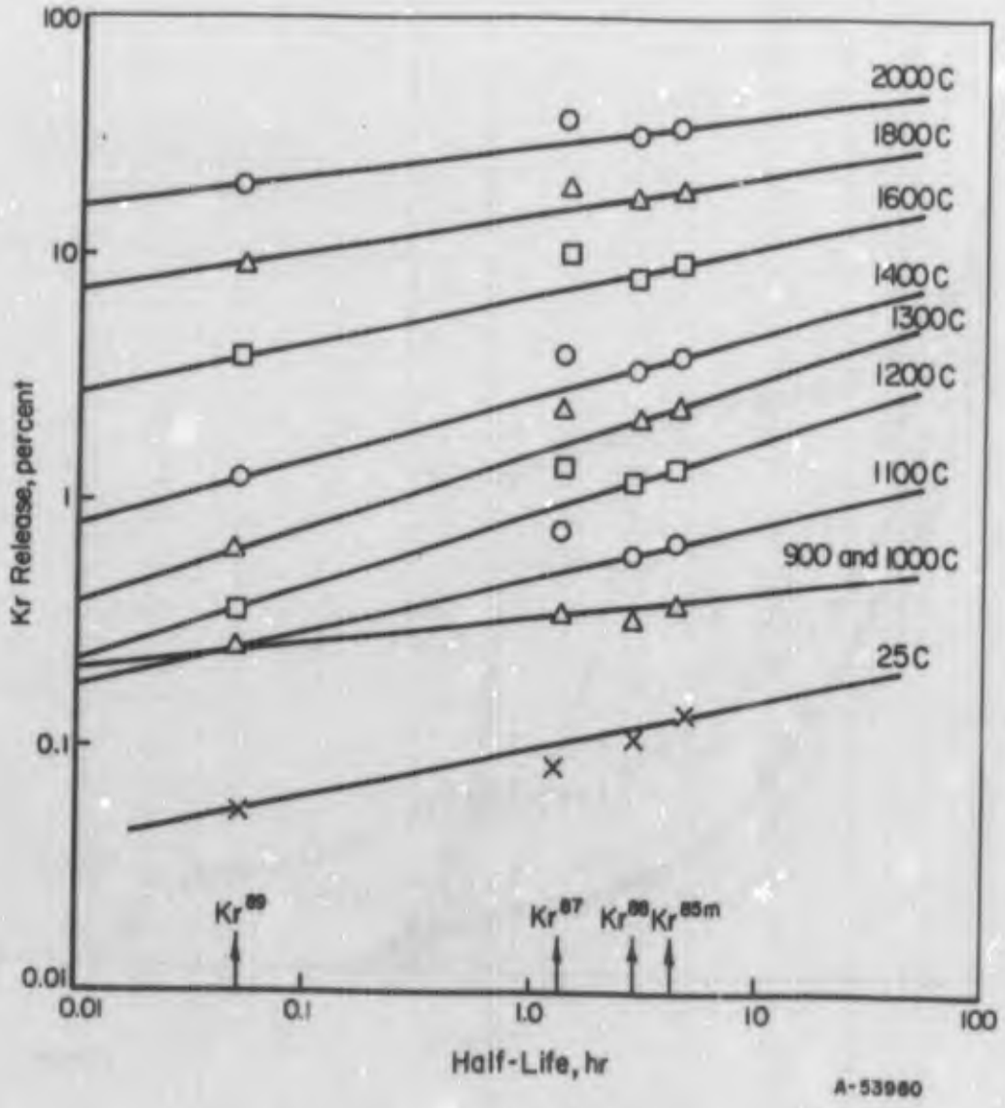


FIGURE 11. STEADY-STATE KRYPTON RELEASE FROM UNCOATED UC DISPERSED IN A GRAPHITE MATRIX(40)

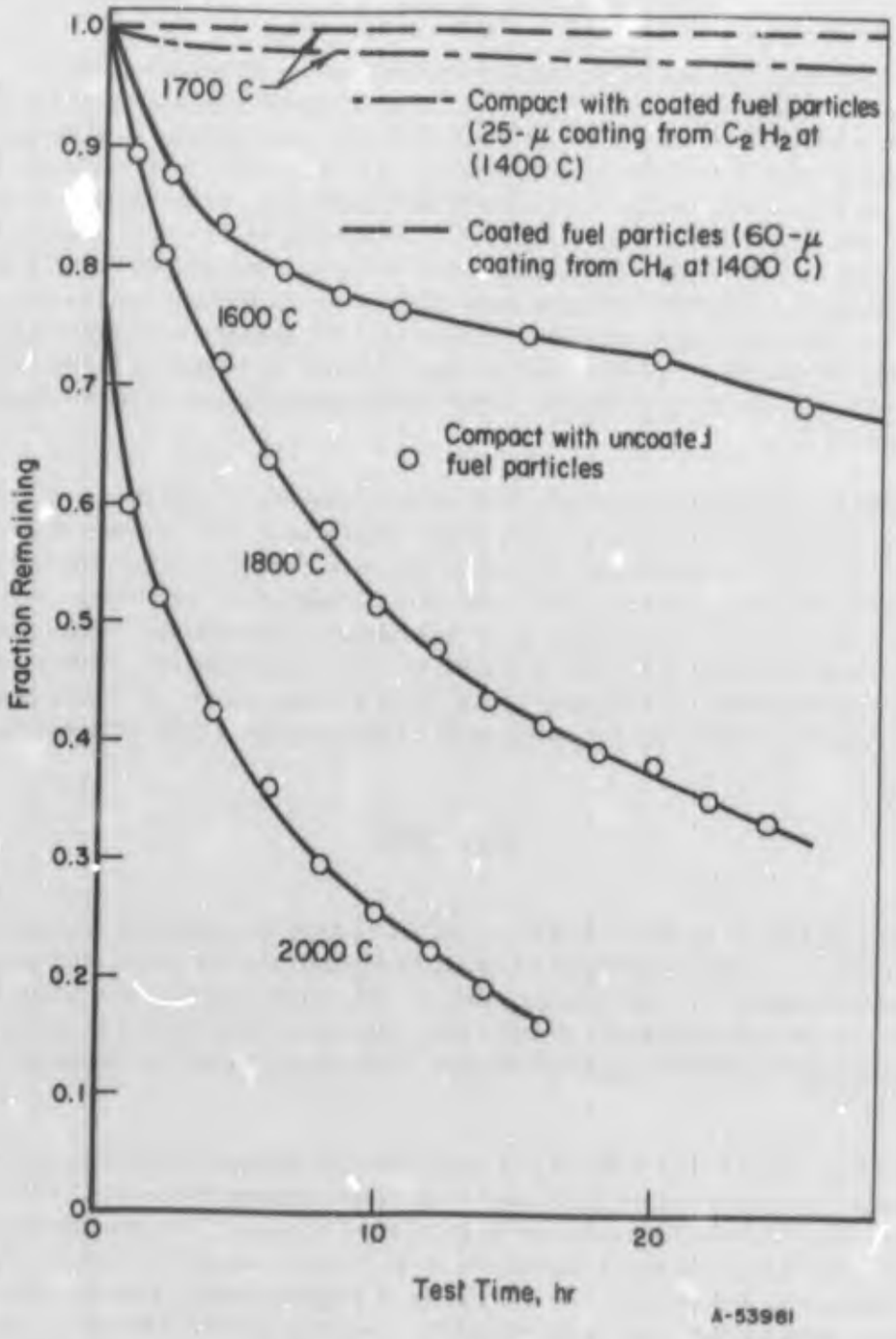


FIGURE 12. RELEASE OF Xe^{133} FROM UNCOATED AND COATED UC PARTICLES DISPERSED IN A GRAPHITE MATRIX(38)

METALLIC MATERIALS

Beryllium

Beryllium was chosen as a matrix material by the Australians mainly because of their desire to have a self-moderating fuel. Hot-pressed specimens of $(U, Th)Be_{13}$ dispersed in a beryllium matrix were irradiated in a predominantly slow-neutron flux, at temperatures of 435 to 530 C and burnups of 4 to 11 x 10^{19} fissions/cm^3.(41) Some specimens contained coarse (100- to 150-μ) fuel particles, while others contained fine (2- to 50-μ) fuel particles. The changes in both the dimensions and density increased with increasing fuel content. Specimens containing fine fuel dispersions underwent larger dimension and density changes than specimens containing coarse fuel particles at similar burnups and irradiation temperatures. No changes in microstructure were caused by the irradiation. The fission gases released increased with the uranium content and burnup of the specimens. Irradiation performance of these dispersions is given in Table 5.

Further irradiations of slightly hyperstoichiometric $(U, Th)Be_{13}$ in beryllium were performed at higher temperatures and burnups as a continuation in performance evaluation.(42) These specimens contained fuel particles varying from 10 to 250 μ in size. Because of the more stringent irradiation conditions, considerably larger dimensional changes and density changes took place. Dispersions containing UBe_{13} dispersed in beryllium melted during irradiation and increased in volume by about 80 percent and released over 30 percent of their fission gases. The microstructure of some specimens shows the appearance of cavities at the grain boundaries.

Aluminum

Aluminum has been found to be a suitable matrix material for dispersion fuels at low temperatures. The advantages of using aluminum are its corrosion resistance to water, excellent thermal conductivity, low neutron cross section and sufficient ductility and strength at the temperatures of interest. The main drawback for using aluminum is its possible reaction with some of the fuel materials if slightly higher temperatures are used.

It has been found that UO_2, U_3O_8, and UAl_3 dispersed in aluminum are stable at the Materials Testing Reactor operating conditions of about 50 to 60 C.(43,44) Irradiation of these dispersions has resulted in increased hardness and tensile strength of the fuel plates. The cause of these increases is believed to be fast-neutron and fission-fragment damage to the matrix. Irradiations at slightly higher temperatures and burnups have been undertaken to develop fuel elements for the Advanced Test Reactor.(45) The fuels that were considered are 35 and 44 weight percent U_3O_8, 32 and 41 weight percent UO_2, and 40 and 54 weight percent UAl_3. In all cases, the matrix material was X-80001 aluminum alloy containing 0.19 weight percent B_4C. In order to improve the performance of the specimens, some of them were coated with Al_2O_3 0.50 to 1.5 mils in thickness. Irradiations took place in the Materials Testing Reactor with the maximum fuel temperatures of 155 to 220 C. Irradiation performances of the different fuel materials dispersed in an aluminum matrix are tabulated in Table 6.

TABLE 5. IRRADIATION PERFORMANCE OF (U Th)Be_{13} DISPERSED IN BERYLLIUM

Sample	Fuel Content, weight percent U	Th	Molar, U:Th ratio	Fuel Volume, %	Diameter, in.	Length, in.	Fuel Particle Size, μ	Density, % T.D.(a)	Cladding Material	Irrad. Temp, C	Irrad. Time, hr	Bumup, 10^{19} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, % Diameter	Density	Volume	Length	Reference
1	6.5	10.2		11	0.23	0.46	100-150		Sodium	520	2400	3.7	0.13	19	0.64	0.55	-0.5		0.4	41
2	6.5	10.2		11	0.23	0.46	100-150		Sodium	465	2400					0.2	-0.1		0.4	41
5	8.2	12.9		16	0.23	0.46	100-150		Sodium	490	2400					0.2	-0.5		0.5	41
6	8.2	12.9		16	0.23	0.46	100-150		Sodium	495	2400	4.8	0.17	25	0.29	0.3	-0.7		0.3	41
7	8.2	12.9		16	0.23	0.46	100-150		Sodium	435	2400	5.1	0.18	26	0.30	0.4	-0.6		0.7	41
8	8.2	12.9		16	0.23	0.46	100-150		Sodium	445	2400	5.1	0.18	26	0.74	0.3	-0.6		0.1	41
9	8.2	12.9		16	0.23	0.46	100-150		Sodium	505	2400	5.3	0.19	27	0.84	0.4	-0.8		0.4	41
10	8.2	12.9		16	0.23	0.46	100-150		Sodium	515	2400	5.3	0.19	27	1.02	0.1	-1.2		0.2	41
17	9.6	11.5		16	0.23	0.46	2-50		Sodium	435	2400					0.6	-3.1		0.6	41
18	9.6	11.5		16	0.23	0.46	2-50		Sodium	450	2400					0.3	-2.5		0.2	41
21	11.2	17.4		24	0.23	0.46	100-150		Sodium	465	2400					0.8	-3.4		1.0	41
22	11.2	17.4		24	0.23	0.46	100-150		Sodium	505	2400	7.5	0.27	39	0.98	0.3	-1.1		0.3	41
25	17.4	27.1		46	0.23	0.46	100-150		Sodium	520	2400					1.35	-5.1		1.3	41
26	17.4	27.1		46	0.23	0.46	100-150		Sodium	530	2400	11.4	0.41	59	1.85	1.8	-4.6		1.3	41
29	16.5	4.7		16	0.23	0.46	100-150		Sodium	510	2400	8.4	0.30	43	1.24	0.6	-4.7		0.6	41
30	16.5	4.7		16	0.23	0.46	100-150		Sodium	500	2400					0.8	-1.2		0.7	41
41			1:2	20	0.19-0.23	0.51	100	96.1	Sodium	450-600	3400	18	0.461	66	0.65	0.9		3.5	1.7	42
42			1:2	20	0.19-0.23	0.51	100	97	Sodium	500-660	3400	16	0.410	59	0.6	1.0		3.8	1.8	42
45			1:2	35	0.19-0.23	0.51	100	95.5	Sodium	350-500	3400	32	0.820	117	8.6	3.1		11.5	5.3	42
46			1:2	35	0.19-0.23	0.51	100	96.3	Sodium	495-670	3400	26	0.666	95		1.4		5.4	2.6	42
49			1:2	50	0.19-0.23	0.51	100	93.3	Sodium	520-640	3400	35	0.898	129	10.7	4.3		15.9	7.3	42
50			1:2	50	0.19-0.23	0.51	100	92.6	Sodium	440-600	3400	34	0.871	125	13.5	3.4		17.9	11.1	42
53			1:2	35	0.19-0.23	0.51	10	96.3	Sodium	520-650	3400	31	0.795	114	0.55	1.1		4.0	1.8	42
54			1:2	35	0.19-0.23	0.51	10	95.5	Sodium	415-620	3400	25	0.640	92	5.4	0.7		2.9	1.5	42
57			1:0	35	0.19-0.23	0.51	100	96.7	Sodium	600-745	3400	19	0.487	70	31.3	35		86.8	16.8	42
58			1:0	35	0.19-0.23	0.51	100	96.7	Sodium	620-850	3400	16	0.410	59	32.7	32		80.5	16.5	42
61			1:2	50	0.19-0.23	0.51	250	92.6	Sodium	520-635	3400	31	0.795	114	1.3	3.5		17.9	10.9	42
62			1:2	50	0.19-0.23	0.51	250	92.3	Sodium	490-600	3400				1.2	2.8				42

(a) T.D. indicates theoretical density.

TABLE 6. IRRADIATION PERFORMANCE OF VARIOUS FUELS DISPERSED IN ALUMINUM

Sample Designation	Number of Specimens in Sample	Fuel Material	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density, % T.D.(a)	Specimen Shape	Width and Thickness, in.	Length, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Center Temperature, C	Irrad. Time, hr	Burnup, 10^{20} fissions/cm³	Bulk Heat Gen. Rate, kw/in³	Surface Heat Gen. Rate, w/in²	Fission-Gas Release, %	Property Changes Caused by Irradiation, % Thickness	Volume	Length	Poison Material	Poison Content, weight, %	References
E-40	1	UO_2	32	10.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	5.9	6.0	165		No failures			B_4C	0.19	45
E-51	1	UO_2	32	10.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	5.4	5.8	95		No failures			B_4C	0.19	45
E-41	1	UO_2	32	10.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	5.6	6.3	150		No failures			B_4C	0.19	45
E-43	1	UO_2	32	10.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	6.0	6.7	160		No failures			B_4C	0.19	45
E-55	1	UO_2	32	10.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	5.7	6.4	160		No failures			B_4C	0.19	45
E-42	1	UO_2	32	10.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	5.7	6.4	160		No failures			B_4C	0.19	45
E-49	1	UO_2	32	10.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	6.1	7.0	195		No failures			B_4C	0.19	45
E-9	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	155	780	6.1	7.0	195		No failures			B_4C	0.19	45
E-11	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	7.4	8.3	208		No failures			B_4C	0.19	45
E-4	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	7.1	7.9	198		No failures			B_4C	0.19	45
E-8	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	6.8	7.6	190		No failures			B_4C	0.19	45
E-26	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	7.7	8.6	216		No failures			B_4C	0.19	45
E-18	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	7.3	8.2	206		No failures			B_4C	0.19	45
E-19	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	7.5	8.4	210		No failures			B_4C	0.19	45
E-25	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	7.7	8.6	216		No failures			B_4C	0.19	45
E-61	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	160	780	7.8	8.7	218		No failures			B_4C	0.19	45
E-60	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.010	160	780	8.3	9.3	234		No failures			B_4C	0.19	45
E-63	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	160	780	7.9	8.8	220		No failures			B_4C	0.19	45
E-400	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	160	780	7.6	8.5	213		No failures			B_4C	0.19	45
E-64	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	160	780	8.6	9.6	241		No failures			B_4C	0.19	45
E-99	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	160	780	8.2	9.2	231		No failures			B_4C	0.19	45
E-101	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	160	780	8.3	9.3	234		No failures			B_4C	0.19	45
E-102	1	UAl_3	40	20.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	160	780	5.3	5.9	148		No failures			B_4C	0.19	45
E-520	1	UO_2	41	14.7	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	160	780	8.7	9.7	243		No failures			B_4C	0.19	45
E-515	1	UO_2	41	14.7	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	220	780	12.5	14.0	350		Blistered			B_4C	0.19	45
E-518	1	UO_2	41	14.7	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	220	780	12.5	14.0	350		Blistered			B_4C	0.19	45
E-519	1	UO_2	41	14.7	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	220	780	12.3	13.7	342		Blistered			B_4C	0.19	45
E-28	1	U_3O_8	44	20.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	220	780	12.3	13.7	342		Blistered			B_4C	0.19	45
E-31	1	U_3O_8	44	20.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	208	780	11.0	12.3	309		Blistered			B_4C	0.19	45
E-33	1	U_3O_8	44	20.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	205	780	6.5	7.3	183		Blistered			B_4C	0.19	45
E-34	1	U_3O_8	44	20.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	205	780	10.2	11.4	285		Blistered			B_4C	0.19	45
E-107	1	UAl_3	44	20.4	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	205	780	11.4	12.7	318		Blistered			B_4C	0.19	45
E-508	1	UAl_3	44	20.4	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	205	780	10.4	11.6	290		No failures			B_4C	0.19	45
E-510	1	UAl_3	44	20.4	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	205	780	10.7	12.0	300		No failures			B_4C	0.19	45
E-507	1	UAl_3	44	20.4	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	205	780	11.0	12.3	310		No failures			B_4C	0.19	45
E-12	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	>105	6061	.015	205	780	11.2	12.5	313		No failures			B_4C	0.19	45
E-13	1	U_3O_8	35	14.9	X-8001		Plate	2.30 x .020	2.88	44-149	6061	.015	180	780	2.8									45
E-1	1	UAl_3	50	28.5	6061	91				>105	6061				3.9									45
E-2	1	UAl_3	50	28.5	X-8001	91				>105	X-8001	.016			30.0									45
BMI-41-1	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	Al	.016	<100	1200	33.0				-6.0	6.1	-0.2			46
BMI-41-2	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	Al	.016	<100	1200	3.4				6.0	5.4	-0.1			46
BMI-41-3	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	Al	.016	<100	1200	3.5				1.4	-1.8[b]	3.0			45
BMI-41-4	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	Al	.016	<100	1200	3.5				2.4	-3.0[b]	2.0			45
BMI-41-5	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	Al	.016	<100	1200	2.9				2.4	-2.4[b]	2.0			45
BMI-41-6	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.02" in. diam	Al-Lipon	.017	<100	1200	2.6				1.6	-2.0[b]	2.0			47
BMI-41-7	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	Fibers	.016	<100	1200	2.8				1.3	-1.6[b]	1.0			48
BMI-41-9	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	None	.016	<100	1200	3.7				1.3	-1.3[b]	0.5			49
BMI-41-10	1	U_3O_8	33	11.0	Aluminum		Plate	.018	2.0	.005 in. diam	None	.016	<100	1200	3.7				12.0	-4.0[b]	1.0			49
							Plate	.018	2.0	.005 in. diam	None	.016	<100	1200	3.4				13.0	-5.0[b]	1.0			49
																			10.0	-4.0[b]	0.0			49

(a) T.D. indicates theoretical density.
(b) Change in density.

It was found that fuel specimens containing UAl_3 performed considerably better than those containing UO_2 and U_3O_8. This is attributed to a reaction between the uranium oxides and the aluminum matrix at these temperatures, which causes an increase of the fuel volume and consequently causes specimen swelling (Figure 13). Previous studies had indicated that a reaction would not be expected at the operating temperatures, but the results indicate that irradiation might decrease the temperature at which reaction between uranium oxides and aluminum takes place. Even with the reaction between fuel and matrix, it was found that the UO_2- and U_3O_8-containing specimens remained dimensionally stable to burnups of 6 to 7 x 10^{20} fissions/cm^3. However, after 11 x 10^{20} fissions/cm^3, all of the oxide-containing fuels had blistered, while the aluminum-UAl_3 dispersions showed no evidence of failure. Metallographic examination showed that fission-gas bubbles in the uranium oxides accompanied reactions with the matrix (Figure 13).

Two samples of aluminum-50 weight percent UAl_3 dispersions were examined after they had been irradiated to burnups of 35 x 10^{20} fissions/cm^3 of fuel in the MTR.[46] Examination of the plates showed that they had undergone a volume increase of 5.4 to 6.1 percent, due almost completely to thickness increases. Metallographic examination revealed that voids were formed between the fuel particles and the matrix.

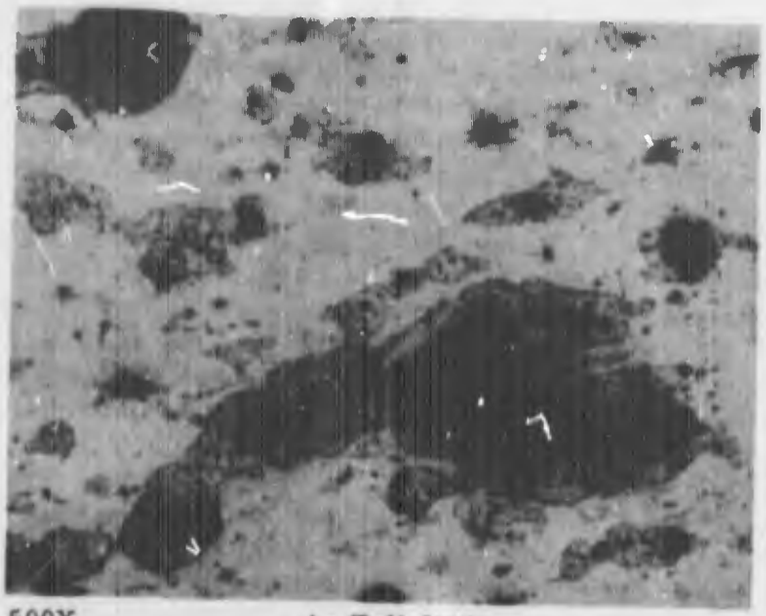
Tensile tests were performed on irradiated dispersions containing UAl_3, UO_2, and U_3O_8 dispersed in an aluminum matrix.[47] The significant effect was the loss of ductility by the specimens.

	Max Elongation, percent
2 x 10^{20} nvt (thermal)	4.5
20	1.0
26	0

Tensile strengths showed increases of 3 to 125 percent in ultimate strength, with most of those of the tested specimens increasing 30 to 80 percent in strength.

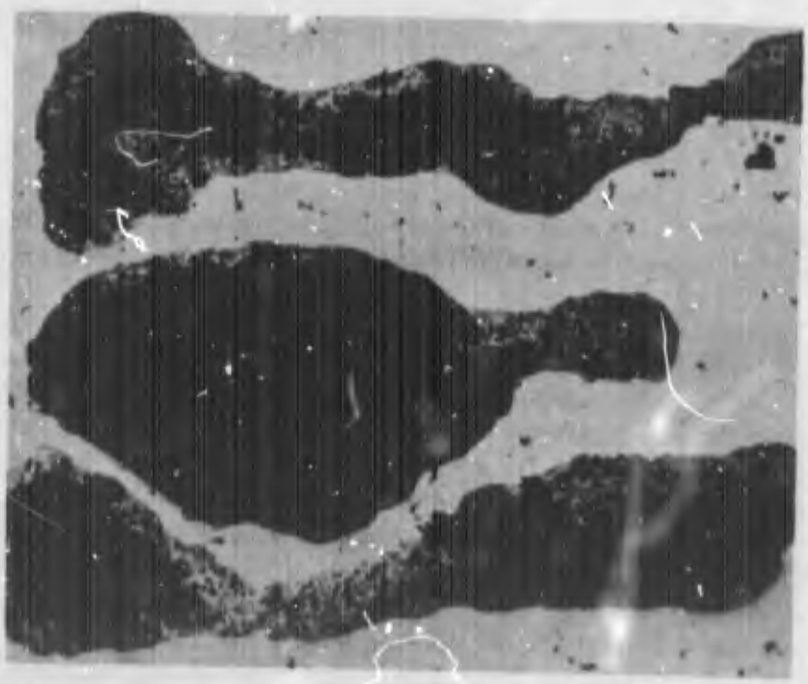
A postirradiation-annealing study was performed on Al-30 weight percent U_3O_8 dispersions[48] that had been irradiated at 55 to 80 C to burnups of 2 to 9.5 x 10^{19} fissions/cm^3. These specimens had undergone no dimensional or density changes as a result of irradiation. No volume changes were found in the dispersions when they were annealed below 500 C. At 550 C, a volume increase of about 4 percent was detected. In comparing the volume change of unirradiated and irradiated dispersions on annealing, it was concluded that the volume increase was due to U_3O_8-aluminum reaction rather than an irradiation-caused effect.

Nine specimens containing 50 volume percent of fueled glass dispersed in 1100 aluminum have been irradiated.[49] The fueled glass contained 50 weight percent U_3O_8, and was dispersed in aluminum as 0.006-inch-diameter fibers. Temperature of the specimens was maintained at about 100 C during irradiation, and the maximum burnup was 3.7 x 10^{20} fissions/cm^3 of fuel, the fast-flux accumulation being 1.8 to 2.6 x 10^{20} nvt. The specimens underwent some dimensional changes, with the unclad specimens exhibiting larger dimensional increases. Fission-gas release from the specimens was less than 1 percent. Tensile tests performed on the specimens indicated that tensile strength had increased to a degree comparable with the increase in strength of the individual matrix and cladding material observed in other studies. No changes in ductility were noted.



500X As Polished

a. Aluminum-24 Weight Percent UO_2 Dispersion



b. Aluminum-35 Weight Percent U_3O_8 Dispersion

FIGURE 13. INTERACTION BETWEEN URANIUM OXIDES AND THE ALUMINUM MATRIX(45)

Iron-Base Alloys

Stainless steel has become the most commonly used matrix material for dispersion fuels. The main advantages offered by stainless steel are its good corrosion resistance in most cooling media, good strength and ductility at moderately high temperatures, fairly low cost, and the large amount of knowledge of fabrication techniques and other pertinent properties. The only mentionable disadvantage for stainless steel appears to be its low thermal conductivity compared to other metals; however, its thermal conductivity is considerably better than those of the fuel materials that are usually dispersed in it.

Iron

The British have irradiated a dispersion of 18.7 weight percent UO_2 in Swedish iron.[50] The fuel rods contained 50- to 200-μ angular UO_2 particles and were extruded to 93 percent of theoretical density. These specimens were irradiated at surface temperatures of 650 C, center temperatures being 750 C. The burnup of the specimens was from 3.5 to 4.9 x 10^{20} fissions/cm^3. No dimensional changes were noted in the specimens after irradiation and the density remained the same. Metallographic examination of the fuel after irradiation showed that bubble formation had taken place in the UO_2 and that the swelling of UO_2 had been accommodated by the initial gaps at the fuel-matrix interfaces. A fission-fragment-damaged zone was also found in the iron matrix.

Some of the irradiated specimens were annealed for 48 hours in the 900 to 1200 C range. The specimens decreased in density as follows: 900 C - 0.2 percent, 1000 C - 2.0 percent, 1050 C - 2.8 percent, and 1175 C - 4.0 percent. The decrease in density is attributed to fission-gas precipitation in the damaged matrix and to swelling of the UO_2.

Ferritic Stainless Steel

A test at Knolls Atomic Power Laboratory was performed to determine the performance of UO_2 particles dispersed in ferritic stainless steel and to compare it to similar dispersions in austenitic stainless steel.[51] The ferritic stainless steel was chosen because of its lower thermal expansion and higher thermal conductivity. The main drawback to using ferritic stainless steels is that irradiation reduces the ductile-brittle transition temperature of the steel, making it susceptible to brittle failure. A ferritic stainless steel-25 weight percent UO_2 specimen was irradiated to a burnup of 5.4 x 10^{20} fissions/cm^3. Examination of the specimen indicated that less sintering and fission-gas-bubble formation had taken place in the ferrite matrix that would be expected in an austenitic stainless steel matrix. This improved performance is attributed to lower central temperatures, which were achieved through higher thermal conductivity of the ferritic stainless steel.

Austenitic Stainless Steel

The early work in stainless steel-UO_2 dispersions has been summarized by an Oak Ridge report.[52] Most of the material given below is taken from that report; however, the reader can refer to the original reports for additional information.

Battelle Memorial Institute and Pratt & Whitney conducted an irradiation program consisting of 112 thin plate specimens containing 30 weight percent UO_2 dispersed in stainless steel. About half the specimens had UO_2 particles smaller than 3 μ, while the other half of the specimens had UO_2 particles in the 15- to 44-μ range. Ninety-six specimens that had 0.013-inch-thick fuel sections were irradiated at temperatures of 175 to 580 C and burnups ranging from 1.1 to 8.9 x 10^{20} fissions/cm^3. None of these specimens failed. However, all eight similar samples, which were irradiated at 965 C to a burnup of 3.9 x 10^{20} fissions/cm^3, failed. Of eight specimens irradiated at 760 C to a burnup of 2.9 x 10^{20} fissions/cm^3, two failed. The specimens that failed contained UO_2 particles that were less than 3 μ, indicating that larger UO_2 particles made for a stabler dispersion. The irradiation performances of all dispersions containing stainless steel as a matrix material are given in Table 7.

An irradiation program carried out by Nuclear Development Associates, Inc., contained four pins and twelve plates. All plates contained 30 weight percent UO_2 dispersed and were clad in 316 stainless steel; the pins had the same fuel loading, but the matrix and cladding material was 310 stainless steel. The plates with a 0.030-inch-thick fuel core were irradiated at temperatures of 650 to 850 C, with the maximum burnup being 1 x 10^{20} fissions/cm^3. None of the plates failed as a result of the irradiation. The 0.080-inch-diameter pins were irradiated at a maximum surface temperature of 630 C and a burnup of 9.4 x 10^{20} fissions/cm^3. One of these irradiated pins warped and showed surface eruptions.

Twelve specimens were irradiated by GEANP. These plates contained 30 weight percent UO_2 and were dispersed and clad in 310 stainless steel. The irradiation temperatures for these specimens varied from 845 to 970 C; the burnup was 1.1 to 3.5 x 10^{19} fissions/cm^3. The only failure occurred in the specimen that had received the lowest burnup and had operated at 870 C.

A total of 97 specimens were irradiated by Knolls Atomic Power Laboratory. These specimens contained 5 to 30 weight percent UO_2 dispersed in stainless steel. The UO_2 particles were of various sizes and the fuel elements were made by various fabrication techniques. The specimens were irradiated at surface temperatures of 150 to 1000 C, with the burnup being from 2 to 32 x 10^{20} fissions/cm^3 of fuel. It was found that pin-like specimens could be irradiated at 650 C to burnups of 10 x 10^{20} fissions/cm^3 of the fuel and remain stable. The difference in fabrication techniques made no difference if the UO_2 particles were in the 20- to 200-μ range.

The platelike fuel elements retained their dimensions and fission gases if irradiated below 700 C. It was found that plates would remain stable when irradiated at 1000 C provided the burnup was less than 2 x 10^{20} fissions/cm^3 of fuel. At surface temperatures of 550 C, the specimens could be irradiated to burnups of up to 32 x 10^{20} fissions/cm^3 of fuel without dimensional changes.

A total of 56 UO_2-stainless steel dispersions were fabricated and irradiated by Oak Ridge National Laboratory. The plates contained from 17.9 to 26.9 weight percent UO_2 and were prepared by three rolling methods. Some of the dispersions contained 0.39 weight percent of natural B_4C as a burnable poison. These specimens were irradiated at low temperatures, the maximum being 100 C, to burnups up to 2.1×10^{20} fissions/cm^3. None of the specimens failed or showed dimensional changes.

A study was undertaken at Battelle to compare the performance of spherical and hydrothermal UO_2 dispersed in 318 stainless steel.(53) The specimens contained 30 weight percent UO_2 and were irradiated at centerline temperatures of 425 to 870 C, with the burnups being 6 to 9×10^{20} fissions/cm^3 of fuel. As a result of irradiation, the specimens underwent thickness increases of 2 to 2.6 percent, while the density decrease ranged from 0.4 to 0.8 percent. There was no apparent difference in the performance of the spherical and hydrothermal UO_2 particles dispersed in stainless steel. Metallographic examination of the irradiated specimens showed the presence of fission-gas bubbles in the UO_2 particles.

A fuel element consisting of 16 stainless steel-25 weight percent UO_2 dispersions was irradiated by Atomics International.(54) The specimens experienced a maximum temperature of 410 C and a burnup of 2.4×10^{20} fissions/cm^3. No measurable dimensional changes of the specimens took place as a result of the irradiation.

An irradiation program involving 62 specimens contained in 10 capsules was undertaken to develop a core for SM-2 (Army Reactor).(55) These specimens contained from 24 to 38 weight percent UO_2 dispersed in Type 347 stainless steel. Also included in the matrix were various amounts of ZrB_2, NbB_2, B_4C, and boron as burnable poisons.

Specimens in the first four capsules were irradiated at surface temperatures of 370 C to a maximum burnup of 20×10^{20} fissions/cm^3 of fuel.(55,56) It was found that specimens that were irradiated at 370 C performed satisfactorily. No difference in performance was found between the spherical or hydrothermal specimens. Also, the performance of the various burnable-poison materials was about the same. Any specimen failures and blistering can be attributed to specimen overheating during irradiation.

Metallographic examination of the specimens revealed that porosity had developed in the UO_2 particles, the amount of porosity increasing with irradiation temperature and burnup. A fission-fragment-damaged zone was detected in the stainless steel matrix which surrounds the UO_2 particles (Figure 14).

Since there was some uncertainty as to the irradiation temperature of the specimens, a postirradiation anneal was performed on four specimens.(55) Four specimens were heated at 370 C for 1 week, after which they were examined. It was found that the specimens had not undergone any dimensional or density changes as a result of the anneal. No fission-gas release from the specimens was noted, indicating that the cladding had maintained its integrity.

Specimens in the other six capsules were irradiated to a higher burnup at somewhat higher temperatures.(57) After examination of the specimens it was concluded that their performance was adequate at 315 to 370 C to burnups of 40×10^{20} fissions/cm^3 of fuel. The type of burnable poison used in the dispersion seemed to have no

TABLE 7. IRRADITION PERFORMANCE OF VARIOUS FUEL

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density, % T.D.	Specimen Shape	Diam. or Thickness, in.	Length, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature, Surface
												Uranium Dioxide
PW-1-2	8	30	24	18,9,73[a]		Plate	0.170 x 0.08	0.57	3-44	347SS	0.006	175
PW-1-4	8	30	24	18,9,73[a]		Plate	0.170 x 0.08	0.57	3-44	347SS	0.006	370
PW-1-5	8	30	24	18,9,73[a]		Plate	0.170 x 0.08	0.57	3-44	347SS	0.006	290
PW-1-C	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	345
PW-1-7	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	345
PW-18	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	370
PW-9	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	455
PW-10	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	345
PW-11	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	430
PW-12	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	405
PW-13	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	580
SP-2	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	760
SP-3	8	30	24	18,9,73[a]		Plate	0.175 x 0.12	0.57	3-44	347SS	0.004	960
1-2	2	30	24	310SS		Plate	0.24 x 0.03	1.49		310SS	0.005	680
1-3	2	30	24	310SS		Plate	0.24 x 0.03	1.49		310SS	0.005	770
1-4	2	30	24	310SS		Plate	0.24 x 0.03	1.49		310SS	0.005	650
1-5	2	30	24	310SS		Plate	0.24 x 0.03	1.49		310SS	0.005	695
1-6	2	30	24	310SS		Plate	0.24 x 0.03	1.49		310SS	0.005	800
1-9	2	30	24	310SS		Plate	0.24 x 0.03	1.49		310SS	0.005	850
1-1	2	30	24	316SS		Rod	0.070	2.74		216SS	0.005	450
1-7	2	30	24	316SS		Rod	0.070	2.74		216SS	0.005	625
3B	1	30	24	310SS		Plate	0.74 x 0.004	1.75		310SS	0.004	900
3C	1	30	24	310SS		Plate	0.74 x 0.004	1.75		310SS	0.004	970
3D	1	30	24	310SS		Plate	0.74 x 0.004	1.75		310SS	0.004	900
3E	1	30	24	310SS		Plate	0.74 x 0.004	1.75		310SS	0.004	870
1B	2	30	24	310SS		Plate	0.74 x 0.004	1.75		310SS	0.004	845
1C	2	30	24	310SS		Plate	0.74 x 0.004	1.75		310SS	0.004	900
1D	4	30	24	310SS		Plate	0.74 x 0.004	1.75		310SS	0.004	900
LST-23	2	30	23.5	Fe		Plate			>3	347SS	0.006	150-175
LST-24	2	30	23.5	Fe					>3	347SS	0.006	150-175
KAPL-14-5	2	30	24	18,9,73[a]		Plate			>3	347SS	0.006	150-175
KAPL-14-6	2	30	24	18,9,73[a]		Plate			>3	347SS	0.006	350
	2	10	7.5	18,9,73[a]		Plate			>3	347SS	0.006	350
KAPL-14-9	2	5	4	23,17,60[a]		Plate			10-44	310SS	0.004	700
	2	10	7.5	23,17,60[a]		Plate			10-44	310SS	0.004	700
	2	20	15.5	23,17,60[a]		Plate			10-44	310SS	0.004	700
	2	30	24	23,17,60[a]		Plate			10-44	310SS	0.004	700
KAPL-14-12	2	5	4	23,17,60[a]		Plate			10-44	310SS	0.004	1000
	2	10	7.5	23,17,60[a]		Plate			10-44	310SS	0.004	1000
	2	20	15.5	23,17,60[a]		Plate			10-44	310SS	0.004	1000
	2	30	24	23,17,60[a]		Plate			10-44	310SS	0.004	1000
KAPL-1-14	2	5	4	23,17,60[a]		Plate			10-44	310SS	0.004	500-550
	2	10	7.5	23,17,60[a]		Plate			10-44	310SS	0.004	500-550
	2	30	24	23,17,60[a]		Plate			10-44	310SS	0.004	500-550

MATERIALS DISPERSED IN AUSTENITIC STAINLESS STEEL

ture, C Center	Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %				Burnable Poison	Poison Content, weight percent	Reference
						Thickness	Density	Volume	Length			
Dispersions												
	500	3.33	5.8	380		No failure						52
	600	8.50	12.3	750		No failure						52
	610	3.4	4.9	313		No failure						52
	425	2.8	5.7	457		No failure						52
	410	3.46	7.3	585		No failure						52
	1040	7.75	6.5	521		No failure						52
	635	7.95	10.9	875		No failure						52
	2550	25.4	8.6	690		No failure						52
		15.1				No failure						52
	4015	21.4	4.6	369		No failure						52
	2415	18.3	6.6	530		No failure						52
	1890	8.3	3.8	305		Two specimens cracked						52
	880	11.1	11.1	890		Eight specimens swelled and cracked						52
	370	2.94	6.9	254		No noticable changes except some warpage						52
	420	3.07	6.4	234		No noticable changes except some warpage						52
	370	1.73	4.0	147		No noticable changes except some warpage						52
	380	1.79	4.1	150		No noticable changes except some warpage						52
	360	1.92	4.6	168		No noticable changes except some warpage						52
	370	1.15	2.7	100		No noticable changes except some warpage						52
	380	2.68	6.1	272		No noticable changes except some warpage						52
	270	1.92	6.2	276		No noticable changes except some warpage						52
	100	0.42	3.7	19		No failure						52
	100	0.55	4.8	24		No failure						52
	250	1.0	3.5	18		No failure						52
	120	0.33	2.4	12		No failure						52
	290	0.70	2.1	11		No failure						52
	70	0.37	4.6	23		No failure						52
	100	0.60	5.2	26		No failure						52
	295	1.3	3.8		0	No change in dimensions						52
	605	3.0	4.3		0	No change in dimensions						52
	600	1.9	2.7		0	No change in dimensions						52
	1920	11.5	5.2		0	No change in dimensions						52
	1920	3.5	1.6		0	No change in dimensions						52
	815	2.0	2.1		0	No change in dimensions						52
	815	3.7	4.0		0	No change in dimensions						52
	815	7.6	8.1		0	No change in dimensions						52
	815	9.9	10.6		9	No change in dimensions						52
	1630	2.0	1.1		Released some fission gas	No change in dimensions						52
	1630	3.7	2.0		Released some fission gas	Bubbles on surface						52
	1630	7.6	4.1		Released some fission gas	Bubbles on surface						52
	1630	11.6	6.2		Released some fission gas	Bubbles on surface						52
	6960	5.3			0	No dimensional changes						52
	6960	10			0	No dimensional changes						52
	6960	32			0	No dimensional changes						52

TABLE 7.

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume %	Matrix Material	Density, % T.D.	Specimen Shape	Diam. or Thickness, in.	Length, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature Surface
KAPL 1L-46	5	25	20	18,12,70	94-98	Rod	0.140	3.0	105-210	304SS	0.010	570
	5	30	24	18,12,70	94-98	Rod	0.100	3.0	105-210	304SS	0.010	570
KAPL 1L-47	5	25	20	27,73	94-98	Rod	0.140	3.0	105-210	304SS	0.010	400
	5	25	20	27,73	94-98	Rod	0.100	3.0	105-210	304SS	0.010	400
SR-565	1	23	18	347SS		Plate	0.200 x 0.51	27	>44	347SS		440
SR-575	1	23	18	347SS		Plate	0.200 x 0.51	27	>44	347SS		410
KAPL-2-37	1	24.5	19.5	347SS		Plate	0.200 x 0.46	27	44-75	347SS		290
KAPL-2-38	1	24.5	19.5	347SS		Plate	0.200 x 0.46	27	44-75	347SS		430
KAPL 18-36	1	24.5	19.5	347SS		Plate	0.200 x 0.48	27	44-75	347SS		425
KAPL 18-38	1	23	18	347SS		Plate	0.200 x 0.50	12	>44	347SS		290
ORNL-21-I	18	25.8	20.5	304SS		Plate				304LSS	0.005-0.007	85
	4	25.8	20.5	304SS		Plate				304LSS	0.005-0.007	85
	4	17.9	14	304SS		Plate				304LSS	0.005-0.007	85
ORNL-17-II	6	18.8	14.5	304SS		Plate				304LSS	0.008	100
	6	22.2	16.5	304SS		Plate				304LSS	0.010	100
	6	26.9	21.5	304SS		Plate				304LSS	0.012	100
ORNL-21-III	12	17.9	14	304SS		Plate				304LSS	0.005	90
BMI 33-1												
59	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150S	318SS		
64	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150H	318SS		
60	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150S	318SS		
65	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150H	318SS		
BMI 33-2												
67	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150H	318SS		
61	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150S	318SS		
68	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150H	318SS		
62	1	30	24	318SS	99	Plate	0.685 x 0.045	1.465	75-150S	318SS		
OMRE	16	25	20	SS		Plate	2.5 x 0.020	26		SS	0.005	
BMI 32-1												
2-7V	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
1-2-2	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
2-5V	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
2-9	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
5-3	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
5-4	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
5-8V	1	26	20.5	347SS		Plate	0.500 x 0.03	2.0	75-150S	347SS	0.0075	370
BMI 32-2												
2-4-14	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
2-2-2	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
3-3	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
3-9	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
8-13	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
5-5V	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
5-1	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
4-10V	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
BMI-32-3												
3-13	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
3-7V	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
2-3-14	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
1-1	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370

(CONTINUED)

ture, C Center	Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %				Burnable Poison	Poison Content, weight percent	Reference
						Thickness	Density	Volume	Length			
	3400	6-12	3.1	273	0	1						52
	3400	7-14	3.6	230	0	1						52
	2290	25-5.5	2.1	185		No dimensional change						52
	2290	25-5.5	2.1	134								52
	2310	9.3	3.5	633	4.7	Swelled						52
	390	1.3	2.9	525	0	No dimensional change						52
	2580	9.6	3.2	560	0	No dimensional change						52
	375	0.96	2.2	385	0	No dimensional change						52
	2380	8.7	3.2	566	0	2-4						52
	2170	22.2	8.9	1600		12						52
		17.8				No failures						52
		10				No failures						52
		6.9				No failures						52
	400	5.7	12.4			No failures						52
	400	7.3	15.9			No failures						52
	400	6.3	13.7			No failures						52
		3.7-20										52
550-745	2260	8.9	3.4	176	0	2.6	-0.8					
530-725	2260	9.1	3.5	181	0	2.0	-0.7					53
495-720	2260	8.3	3.2	165	0	2.2	-0.7					53
425-605	2260	8.8	3.4	176	0	2.6	-0.5					53
605-850	1470	7.1	4.2	217		2.4	-0.4					
620-880	1470	7.1	4.2	217		2.4	-0.6					53
	1470	6.5	3.8	191		2.2	-0.7					53
535-705	1470	5.9	3.5	181		2.0	-0.4					53
415		2.4				No dimensional changes						53
												54
	3600	17	4.1	142	3.9	Blistered						
	3600					5.3	-1.9		0.07	ZrB_2	0.095	55,56
	3600					6.0	-2.2		0.15	B_4C	0.192	55,56
	3600					2.6	-2.7		0.79	ZrB_2	0.199	55,56
	3600	19.5	4.7	163		4.8	-2.5		0.11	NbB_2	0.209	55,56
	3600	19.2	4.6	160		4.5	-2.5		0.04	–		55,56
		17.5	4.2	143		5.5	-1.5			–		55,56
	3360									ZrB_2	0.240	55,56
	3360	19.8	5.1	177		5.0	-1.4		0.17	B^{10}	0.073	55,56
	3360					6.3	-2.7			B_4C	0.192	55,56
	3360					5.5	-2.9		0.17	–		55,56
	3360					5.4	-2.6			NbB_2	0.209	55,56
	3360	19.4	5.0	173		5.8	-3.1		0.15	NbB_2	0.202	55,56
	3360					6.6	-2.9			ZrB_2	0.199	55,56
	3360	15.9	4.1	140		4.8	-2.4		0.19	B_4C	0.021	55,56
						4.2	-2.2		0.09	ZrB_2	0.192	55,56
	2800	15.4	4.0	163		4.2	-2.0		0.26	NbB_2	0.204	55,56
	2800					4.5	-1.8			ZrB_2	0.095	55,56
	2800	15.9	4.9	167		3.6	-1.7		0.33	B^{10}	0.023	55,56
	2800					3.9	-1.9		0.75	B_4C	0.021	55,56

TABLE 7.

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density, % T.D.	Specimen Shape	Diam. or Thickness, in.	Length, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature, Surface
KAPL-1-13	2	10	7.5	23,17,60[a]		Plate			10-44	310SS	0.004	350
	2	20	15.5	23,17,60[a]		Plate			10-44	310SS	0.004	350
	2	30	24	23,17,60[a]		Plate			10-44	310SS	0.004	350
B-9	1	30	24	18,12,70[a]		Rod	0.090	1.1	44-74	347SS	0.003-0.020	500-550
	1	30	24	18,12,70[a]		Rod	0.160	1.1	44-74	347SS	0.003-0.020	500-550
B-10	1	30	24	18,12,70[a]		Rod	0.090	1.1	44-74	347SS	0.003-0.020	500-55[illegible]
	1	30	24	18,12,70[a]		Rod	0.160	1.1	44-74	347SS	0.003-0.020	500-550
C-1	2	30	24	18,12,70[a]		Rod	0.090	1.1	44-74	347SS	0.003-0.020	400-450
	1	30	24	18,12,70[a]		Rod	0.160	1.1	44-74	347SS	0.003-0.020	400-450
C-2	2	30	24	18,12,70[a]		Rod	0.090	1.1	44-74	347SS	0.003-0.020	400-450
	1	30	24	18,12,70[a]		Rod	0.160	1.1	44-74	347SS	0.003-0.020	400-450
C-3	2	30	24	18,12,70[a]		Rod	0.090	1.1	44-74	347SS	0.003-0.020	400-450
	1	30	24	18,12,70[a]		Rod	0.160	1.1	44-74	347SS	0.003-0.020	400-450
KAPL-18-13	1	30	24	347SS		Rod	0.140	24-27	44-74	347SS	0.010	450
KAPL-18-14	1	30	24	347SS		Rod	0.140	24-27	44-74	347SS	0.010	450
KAPL-18-22	1	23	18	347SS		Rod	0.140	24-27	44-74	347SS	0.010	535
KAPL-18-27	1	23	18	347SS		Rod	0.140	24-27	44-74	347SS	0.010	650
KAPL-18-28	1	23	18	347SS		Rod	0.140	24-27	44-74	347SS	0.010	600
KAPL-18-29	1	23	18	347SS		Rod	0.140	24-27	44-74	347SS	0.010	600
KAPL-18-30	1	23	18	347SS		Rod	0.140	24-27	44-74	347SS	0.010	460
KAPL-18-35	1	23	18	347SS		Rod	0.140	24-27	44-74	347SS	0.010	460
KAPL-18-43	2	24	19	347SS	91.2	Rod	0.140	24-27	105-210	347SS	0.010	450
	1	24	19	347SS	96	Rod	0.140	24-27	>44	347SS	0.010	450
KAPL-18-44	2	24	19	347SS	91.2	Rod	0.140	24-27	105-210	347SS	0.010	500
	1	24	19	347SS	96	Rod	0.140	24-27	>44	347SS	0.010	500
LTS B-19	2	24	19	18,12,70[a]	91	Rod	0.140	1.0	105-210	304SS	0.010	450
	2	24	19	347SS	95	Rod	0.140	1.0	>44	347	0.010	450
LTS B-20	2	24	19	18,12,70[a]	91	Rod	0.140	1.0	105-210	304	0.010	450
	2	24	19	347SS	95	Rod	0.140	1.0	>44	347		450
LTS B-24	1	24	19	18,12,70[a]	91	Rod	0.140	1.5	105-210	304	0.010	450
	1	24	19	347SS	95	Rod	0.140	1.5	>44	347	0.010	450
LTS C-14	1	24	19	18,12,70[a]	91	Rod	0.140	1.5	105-210	304	0.010	450
	1	24	19	347SS	95	Rod	0.140	1.5	>44	347	0.010	450
KAPL 18L-71	1	25	20	18,12,70[a]	94-98	Rod	0.140	17	105-210	304SS	0.010	1100-1300
KAPL 18L-72	1	25	20	18,12,70[a]	94-98	Rod	0.140	12.2	105-210	304SS	0.010	700-900
KAPL 18L-77	1	25	20	18,12,70[a]	94-98	Rod	0.140	20.9	105-210	304SS	0.010	485
	1	25	20	18,12,70[a]	94-98	Rod	0.100	26.2	105-210	304SS	0.010	485
KAPL 18L-78	2	28.5	22.5	18,12,70[a]	94-98	Rod	0.140	20.4	105-210	304SS	0.010	
KAPL 18L-79L	1	25	20	18,12,70	94-98	Rod	0.140	21	105-210	304SS	0.010	570
	1	25	20	18,12,70	94-98	Rod	0.120	26.6	105-210	304SS	0.010	570
KAPL 1L-40	2	25	20	18,12,70	94-98	Rod	0.140	2	105-210	304SS	0.010	400-800
	2	25	20	18,12,70	94-98	Rod	0.140	1.5	105-210	304SS	0.010	400-800
	2	25	20	18,12,70	94-98	Rod	0.140	1.5	105-210	304SS	0.010	400-800
	2	25	20	18,12,70	94-98	Rod	0.140	1.5	105-210	304SS	0.010	400-800
KAPL 1L-41	5	25	20	18,12,70	94-98	Rod	0.140	3.0	105-210	304SS	0.010	400-800
	5	30	24	18,12,70	94-98	Rod	0.100	3.0	105-210	304SS	0.010	400-800
KAPL 1L-43	5	25	20	18,12,70	94-98	Rod	0.140	3.0	105-210	304SS	0.010	400-800
	5	30	24	18,12,70	94-98	Rod	0.100	3.0	105-210	304SS	0.010	400-800

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ture, C Center	Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %				Burnable Poison	Poison Content, weight percent	Reference
						Thickness	Density	Volume	Length			
	3880	3.3			0	No dimensional changes						52
	3880	6.3			0	No dimensional changes						52
	3880	20			0	No dimensional changes						52
	830	5.1	5.3		0	No dimensional changes						52
	830	4.3	4.5	450	0	No dimensional changes						52
	1605	10	5.4		0	No dimensional changes						52
	1605	8.3	4.5	450	0	No dimensional changes						52
	1660	16.5	8.6		0.2			Cracked				52
	1660	14.2	7.4	740	0.2	2		Cracked				52
	2520	20.1	6.9					Cracked				52
	2520	16.5	5.7	570	9.2	5		Cracked				52
	3310	23.6	6.2		0	0.8		Cracked				52
	3310	20.7	5.4	540	0	0.8		Cracked				52
	990	5.3	4.7	414		No dimensional changes						52
	610	2.2	3.1	273		No dimensional changes						52
	1160	2.0	1.5	132		No dimensional changes						52
	2320	9.3	3.5	308		No dimensional changes						52
	1980	8.0	3.5	308		No dimensional changes						52
	1570	5.6	3.1	273	0	No dimensional changes						52
	790	3.0	3.3	290	1.4	No dimensional changes						52
	4490	12.4	2.4	211	0	No dimensional changes						52
	1460	3.8	2.3	202		No dimensional changes						52
	1460	3.8	2.3	202		No dimensional changes						52
	3250	7.5	2.0	176		No dimensional changes						52
	3250	7.5	2.0	176		No dimensional changes						52
	755	1.4	1.6	141		No dimensional changes						52
	755	1.4	1.6	141		No dimensional changes						52
	1830	3.7	1.8	158		No dimensional changes						52
	1830	3.7	1.8	158	4.1	No dimensional changes						52
	1140	2.8	2.1	185	0	No dimensional changes						52
	1140	2.8	2.1	185		No dimensional changes						52
	1140	0.93	0.71	62	0	No dimensional changes						52
	1140	0.93	0.71	62								52
	1240	16.3	11.5	1000	46	Broken						52
	2280	23.6	9.0	790	64	Broken						52
	2640	5.9	1.9	167	0	No dimensional changes						52
		5.9	1.9	122		No dimensional changes						52
	3890	16.6	3.7	326	0	1.0						
	2790	8.9	2.8	256	0	No dimensional change						52
	2790	8.9	2.8	213	0	No dimensional change						52
	2130	5.9	2.4	211								
	3660	12.3	2.9	255	Released some gas	5		Cracked				52
	325	1.0	2.7	238	Released some gas	4		Cracked				52
	1530	4.0	2.3	202		No dimensional change						52
	2660	6-10	3.3	290		Eight specimens cracked						52
	2660	7-12	3.9	250								52
	2460	5-7.5	2.7	237		Eight specimens cracked						52
	2460	6-9	3.2	205								52

TABLE 7.

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density, % T.D.	Specimen Shape	Diam. or Thickness, in.	Length, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature Surface
BMI-32-3												
4-4	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
2-10V	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
1-8V	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
BMI-32-4												
1-12V	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
2-3	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
4-7A	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
3-4	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
2-6V	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
4-11	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
BMI-32-5												
4-5B	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
2-6A	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	370
5-11	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	370
2-13	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	365
5-17B	1	38.4	31.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	510
BMI-32-6												
3-1A	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	260
2-4A	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	305
3-9A	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	320
4-13A	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	330
1-16B	1	38.4	31.5	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	
BMI-32-7												
4-1A	1	24	19	347SS		Plate	0.500 x 0.03	1.0	75-150H	347SS	0.0075	215
3-4A	1	26	20.5	347SS		Plate	0.500 x 0.03	1.0	75-150S	347SS	0.0075	390
3-5C	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	320
3-6V	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	280
2-16B	1	38.4	31.5	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	
3-17B	1	38.4	31.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	
BMI-32-8												
1-1A	1	24	19	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	310
1-4A	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	305
7-13	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	250
2-14A	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	320
1-8A	1	26	20.5	347SS		Plate	0.500 x 0.030	2.0	75-150S	347SS	0.0075	
BMI-32-9												
2-1A	1	24	19	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	700
5-5B	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	490
2-7A	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	430
3-13A	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	605
3-8A	1	26	20.5	347SS		Plate	0.500 x 0.030	2.0	75-150S	347SS	0.0075	
BMI-32-10												
4-3	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	
4-5C	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	230
1-6A	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	370
2-9A	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150H	347SS	0.0075	310
1-12C	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	
3-14A	1	26	20.5	347SS		Plate	0.500 x 0.030	1.0	75-150S	347SS	0.0075	

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ture, C Center	Irrad. Time hr	Burnup, 10^{20} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, % Thickness	Density	Volume	Length	Burnable Poison	Poison Content, weight percent	Reference
	2800											
	2800	15.4	4.8	163	9.6	3.6	-1.8		0.24	–	–	55,56
	2800					Blistered				ZrB_2	0.192	55,56
						Blistered				ZrB_2	0.240	55,56
	2060	17.3	7.3	250		6.4	-2.2			ZrB_2	0.209	55,56
	2060					2.9	-2.1		0.09	–		55,56
	2060					3.6	-1.6		0.05	ZrB_2	0.093	55,56
	2060					5.5	-1.8		0.05	–		55,56
	2060					1.7	-1.4		0.12	ZrB_2	0.213	55,56
	2060	9.1	3.8	130		2.7	-0.9		0.86	NbB_2	0.207	55,56
	4420	25.3	5.0	170		8.9	-4.94	0.07[b]	0.07	ZrB_2	0.197	55,56,57
	4420	24.8	4.9	167		7.5	-4.15	0.10[b]	-0.17	ZrB_2	0.200	55,56,57
	4420	28.3	5.5	187		9.4	-5.04	0.50[b]	-0.05	NbB_2	0.207	55,56,57
	4420	25.1	4.9	167		7.8	-4.45	0.39[b]	0.19	NbB_2	0.203	55,56,57
	4420	37.1	7.3	250		160	-30.9	-0.10[b]	-0.05	ZrB_2	0.350	55,56,57
	4730	19.1	3.5	129		4.1	-2.9	-0.05[b]	-0.09	B_4C	0.048	55,56,57
	4730	21.6	3.9	133		-5.1	-3.31	0.14[b]	0.07	–	–	55,56,57
	4730	21.7	3.9	133		6.4	-3.58	0.05[b]	-0.03	NbB_2	0.222	55,56,57
	4730	22.3	4.1	140		6.7	-3.96	0.37	-0.01	NbB_2	0.204	55,56,57
	4730					Disintegrated				ZrB_2	0.430	55,56,57
	5080	22.4	3.8	130		20.5	-6.36	-0.37[b]	-0.09	B_4C	0.048	55,56,57
	5080	24.9	4.3	146		6.7	-3.68	-0.36[b]	0.06	–	–	55,56,57
	5080	25.5	4.3	149		7.5	-4.44	-0.47[b]	0.05	ZrB_2	0.183	55,56,57
	5080	24.2	4.1	142		7.5	-4.28	0.77[b]	-0.14	ZrB_2	0.213	55,56,57
	508C	39.2	6.7	232		274	-45.9	-6.24[b]	-0.03	ZrB_2	0.430	55,56,57
	5080	36.4	6.2	214		202	-37.7	-1.61[b]	-0.09	ZrB_2	0.350	55,56,57
	4730	21.0	3.8	132		6.6	-3.64	0[b]	0.05	B_4C	0.048	55,56,57
		22.9	4.2	145		5.4	-3.60	0.16[b]	0.10	–	–	55,56,57
		23.7	4.4	152		6.8	-4.04	0.14[b]	0.13	NbB_2	0.202	55,56,57
						6.8	-4.35	0.23[b]	0.11	B^{10}	0.023	55,56,57
		24.9	4.6	160		2.4		0.03[b]		ZrB_2	0.222	55,56,57
	4800	21.8	4.0	138		Disintegrated				B_4C	0.048	55,56,57
		23.9	4.3	149		Disintegrated				ZrB_2	0.197	55,56,57
		23.9	4.3	149		72.2	-18.6	0.05[b]	0	ZrB_2	0.093	55,56,57
		28.4	5.1	177		11.5	-5.8	0.05[b]		NbB_2	0.204	55,56,57
		30.7	5.6	194		58.2	Blistered	0.17[b]		ZrB_2	0.222	55,56,57
	4420	23.2	4.6	156		5.5	-3.65	-0.10[b]	-0.03	–	–	55,56,57
		24.1	4.7	163		5.8	-3.60	0.19[b]	-0.15	ZrB_2	0.183	55,56,57
		23.2	4.6	159		7.4	-4.02	0.10[b]	-0.03	ZrB_2	0.200	55,56,57
		24.4	4.7	163		7.6	-4.34	-0.05[b]		NbB_2	0.222	55,56,57
		25.4	5.0	173		6.0	-3.84	-0.07[b]	-0.03	ZrB_2	0.168	55,56,57
		25.2	5.0	173		6.5	-4.40	0.12[b]	0.06	B^{10}	0.023	55,56,57

TABLE 7.

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density, % T.D.	Specimen Shape	Diam. or Thickness, in.	Length, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature Surface
BMI-28-1												
51	1	30	24	318SS		Plate	0.688 x 0.030	1.47	75-150H	318SS	0.015	815-1030
S-72	18	25	19.5	302B						304L	0.005	
S-79	18	25	19.5	302B						304L		
S-81	18	25	19.5	302B						304L		
BMI-46-2												
5	1	33.2	26.5	347SS	93	Plate	0.35 x 0.102	1.5	105-150S	347SS	0.007	540
6	1	33.2	26.5	347SS	93	Plate	0.35 x 0.102	1.5	105-150S	347SS	0.007	
BMI-46-4												
1	1	33.2	26.5	347SS	93	Plate	0.35 x 0.102	1.5	105-150S	347SS	0.007	480
10	1	33.2	26.5	347SS	93	Plate	0.35 x 0.102	1.5	105-150S	347SS	0.007	510
BMI-46-5												
8	1	33.2	26.5	347SS	93	Plate	0.35 x 0.102	1.5	105-150S	347SS	0.007	540
9	1	33.2	26.5	347SS	93	Plate	0.35 x 0.102	1.5	105-150S	347SS	0.007	540
S-14	1	58	40	316L		Rod	0.235			316L	0.020	
S-15	1	58	50	316L		Plate	0.360 x 0.048	1.97	250-700	316L	0.018	645-655
S-16	1	48	40	316L		Plate	0.360 x 0.048	1.97	250-700	316L	0.018	650-660
S-17	1	37	30	316L		Plate	0.360 x 0.048	1.97	250-700	316L	0.018	655-665
S-18	1	58	50	316L		Plate	0.360 x 0.048	1.97	250-700	316L	0.018	620-630
S-19	1	48	40	316L		Plate	0.360 x 0.048	1.97	250-700	316L	0.018	635-645
												Uranium Carbide
BMI-18-1												
4-A	1	24	15.8	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
4-B	1	24	15.8	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
BMI-18-2												
4-C	1	24	15.8	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
4-E	1	24	15.8	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
BMI-18-3												
(2)	1	24	15.8	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
												Uranium Nitride
BMI-18-1												
3-E	1	24	15	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
4-B	1	24	15	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
BMI-18-2												
4-C	1	24	15	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
4-D	1	24	15	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
BMI-18-3												
(7)	1	24	15	318SS		Plate	0.675 x 0.030	1.5		318SS	0.008	
BMI-28-1												
55	1	28	17.9	318SS		Plate	0.688 x 0.030	1.47	75-150	318SS	0.015	860-900
56	1	28	17.9	318SS		Plate	0.688 x 0.030	1.47	75-150	318SS	0.015	865-915
57	1	28	17.9	318SS		Plate	0.688 x 0.030	1.47	75-150	318SS	0.015	795-865
BMI-32-5												
2-15B	1	34	22.5	347SS		Plate	0.45 x 0.030	1.0	75-150	347SS		510
3-15B	1	34	22.5	347SS		Plate	0.45 x 0.030	1.0	75-150	347SS		

(CONTINUED)

ture, C Center	Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, % Thickness	Density	Volume	Length	Burnable Poison	Poison Content, weight percent	Reference
	1600	21	1.0	35			-1.2					67
		6-15				No significant dimensional changes				B_4C	0.13	60,61
		17-23				No significant dimensional changes				B_4C	0.13	60,51
		25				10 (Max)				B_4C	0.13	60,71
	10400	17	1.7	185			-13					62
	10400					Failed						62
	2300	3.6	1.7	185			-0.61					62
	2300	3.9	1.7	185			-0.39					62
	10000	20	2.0	226			-8.3					62
	10000	20	2.0	226			-6.1					62
670		6.2				No dimensional changes						63
	4200	13.8	3.5	182				6.3				63
	4200	11.1	3.4	176				5.9				63
	4200	8.9	2.5	130				2.0				63
	4200	9.9	2.4	124				4.3				63
	4200	4.8	1.35	70				0.5				63
Dispersions												
930	860	2.6	2.6	93			-0.37					64,65,66
980	860	2.6	2.6	93			-6.1					64,65,66
870	430	1.3	2.6	93			-0.08					64,65,66
810	430	1.3	2.6	93			-0.28					64,65,66
795	1720	5.2	2.6	93								
Dispersions												
870	860	2.6	2.6	93			-0.37					64,65,66
980	860	2.6	2.6	93			-2.2					64,65,66
810	430	1.3	2.6	93			-0.28					64,65,66
940	430	1.3	2.6	93			0					64,65,66
795	1720	5.2	2.6	93								
	1600	1.9	1.0	35			-1.3					67
	1600	2.2	1.2	42			-1.5					67
	1600	1.7	0.93	33			-0.6					67
	4420	35.7	7.0	242		65.9	-23		-0.10	ZrB_2	0.315	55,56,57
	4730					Disintegrated				ZrB_2	0.315	55,56,57

TABLE 7.

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density, % T.D.	Specimen Shape	Diam. or Thickness, in.	Length, in.	Fuel Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature, Surface
												Plutonium Oxide
1	1	57	50	316L		Rod	0.235			316L	0.020	
2	1	57	50	316L		Rod	0.235			316L	0.020	
3	1	57	50	316L		Rod	0.235			316L	0.020	
4	1	57	50	316L		Rod	0.235			316L	0.020	
5	1	57	50	316L		Rod	0.235			316L	0.020	
6	1	57	50	316L		Rod	0.235			316L	0.020	
7	1	57	50	316L		Rod	0.235			316L	0.020	
8	1	47	40	316L		Rod	0.235			316L	0.020	
9	1	47	40	316L		Rod	0.235			316L	0.020	
10	1	47	40	316L		Rod	0.235			316L	0.020	
11	1	57	50	316L		Rod	0.235			316L	0.020	
12	1	47	40	316L		Rod	0.235			316L	0.020	
13	1	47	40	316L		Rod	0.235			316L	0.020	
20	1	57	50	316L	84-88	Rod	0.098		100-350	316L	0.020	585-595
21	1	57	50	316L	84-88	Rod	0.098		100-350	316L	0.020	585-595
22	1	36	30	316L	84-88	Rod	0.098		100-350	316L	0.020	585-595
23	1	47	40	316L	84-88	Rod	0.098		100-350	316L	0.020	585-595
24	1	47	40	316L	84-88	Rod	0.098		100-350	316L		635-645
25	1	36	30	316L	84-88	Rod	0.098		100-350	316L	0.020	635-645
26	1	36	30	316L	84-88	Rod	0.098		100-350	316L	0.020	635-645
27	1	47	40	316L	84-88	Rod	0.098		100-350	316L	0.020	635-645

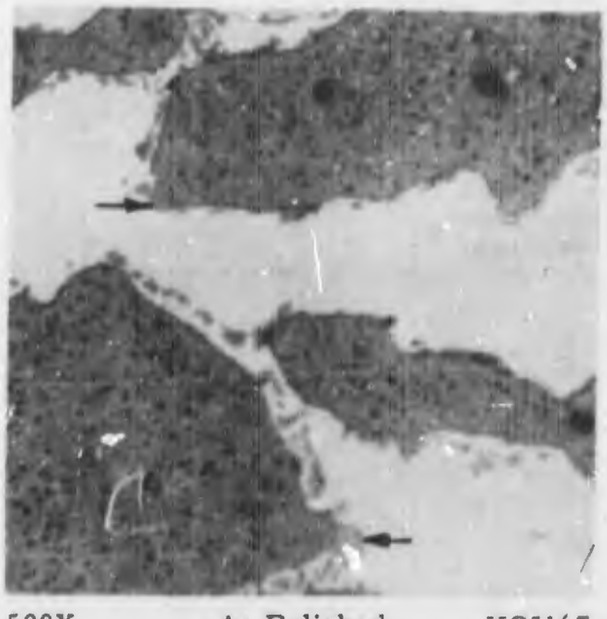
(a) The three numbers refer to percent elemental compositions in this order: chromium, nickel, iron.
(b) Change in width rather than volume.
H Hydrothermal particle
S Spherical particle

(CONTINUED)

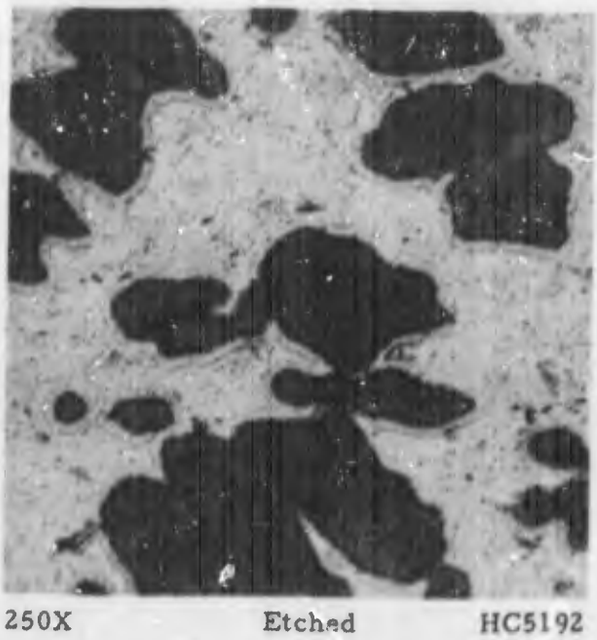
ture, C Center	Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, % Thickness	Density	Volume	Length	Burnable Poison	Poison Content, weight percent	Reference
Dispersions												
690		0.28			5.4	No dimensional changes						63
500		0.56			6.7	No dimensional changes						63
		0.04										63
		0.72			3.7	No dimensional changes						63
900		1.8				No dimensional changes						63
700-800		8.8			29							63
480		0.17				No dimensional changes						63
540-700		2.4			3.2	No dimensional changes						63
605		0.38				No dimensional changes						63
700		5.7			2.5							63
250-400		2.7				No dimensional changes						63
600-800		8.8			100							63
740-840		8.8										63
	4200				3.9	No dimensional changes						63
	4200					No dimensional changes						63
	4200				3.9	No dimensional changes						63
	4200				2.4	No dimensional changes						63
	4200				3.6	No dimensional changes						63
	4200				8.6	No dimensional changes						63
	4200				5.3	No dimensional changes						63
	4200				5.2	No dimensional changes						63

500X As Polished RM12750

a. Unirradiated



500X As Polished HC5167



250X Etched HC5192

b. Irradiated to a Burnup of 20×10^{20} Fissions/Cm^3

FIGURE 14. COMPARISON OF THE PRE- AND POSTIRRADIATION MICROSTRUCTURE OF 26 WEIGHT PERCENT UO_2 DISPERSED IN 347 STAINLESS STEEL

SM-2 Core, Army Reactor. Note the fission-gas bubbles in UO_2 and the recoil zone in stainless steel that surrounds the UO_2. Sintering of the cracks in UO_2 has occurred during irradiation.

effect on the specimen performance. As before, the failures were attributed to high operating temperatures. It was found that cracking in the matrix took place when the irradiation temperature was above 540 C and the burnup reached 30 x 10^{20} fissions/cm^3 of fuel.

Metallographic examination of the irradiated specimens showed bubble formation in the UO_2 particles and a fission-fragment-damaged zone in the stainless steel that surrounded the UO_2 particles. A white phase, which was found in the UO_2 particles, was not identified. However, recently, similar particles found in UO_2 fuel elements have been identified as fission-product-stabilized metallic uranium. (58,59)

Postirradiation annealing of selected specimens showed that no blistering or density changes took place after annealing at 315 C. Minor blisters appeared after annealing at 370 C, but no gross density changes took place even after annealing at 540 C.

Three SM-1 fuel elements, each element containing 18 fuel plates, have been irradiated and examined. (60,61) The fuel plates were stainless steel-24 weight percent UO_2 dispersions containing 0.13 weight percent B_4C as a burnable poison. These fuel plates underwent burnups up to 25 x 10^{20} fissions/cm^3 of fuel. The maximum dimensional change was a 10 percent increase in thickness, but considerable amounts of rippling on the surface occurred. The amount of rippling was found to increase with the amount of in-pile time. Metallographic examinations revealed bubble formation in the UO_2. No matrix cracking was detected, although some cracking observed in the clad was attributed to chloride stress corrosion.

Six fuel plates containing stainless steel-33 weight percent UO_2 dispersions were tested as possible fuel for Core B of the Enrico Fermi Fast Breeder Reactor. (62) The fuel consisted of spherical UO_2 particles, 105 to 150 μ in size, dispersed in Type 347 stainless steel. These specimens were irradiated at surface temperatures of 480 to 540 C to burnups of 3.6 to 20 x 10^{20} fissions/cm^3. Swelling of the specimens did take place, and was found to be proportional to irradiation temperature and burnup. It was concluded that the maximum allowable burnup for the fuel at 480 C was about 16 x 10^{20} fissions/cm^3. Metallographic examination of the irradiated specimen showed a microstructure similar to those of other stainless steel-UO_2 dispersions: bubble formation and white phase in UO_2, along with a recoil-damaged zone in the stainless steel that surrounds the UO_2.

A convenient method of predicting the performance of stainless steel-UO_2 dispersions at various temperatures and burnups is illustrated in Figure 15. In this graph, a failed specimen is one that ruptures, blisters, or swells enough (usually greater than 10 percent thickness change) to cause plugging of coolant channels in the projected reactor. From Figure 15 it can be observed that continuous improvement of the irradiation performance of the stainless steel-UO_2 dispersions has been obtained in the last few years by improved fabrication techniques.

Five roll-bonded stainless steel-UO_2 plates and one pin containing varying UO_2 contents were irradiated by the British. (63) These specimens were irradiated at cladding temperatures of 620 to 665 C, with the burnup varying from 5 to 14 x 10^{20} fissions/cm^3 of fuel. The volume changes undergone by the plates as a result of the irradiation were found to be dependent on irradiation temperature and burnup. The pin underwent no dimensional changes as a result of irradiation.

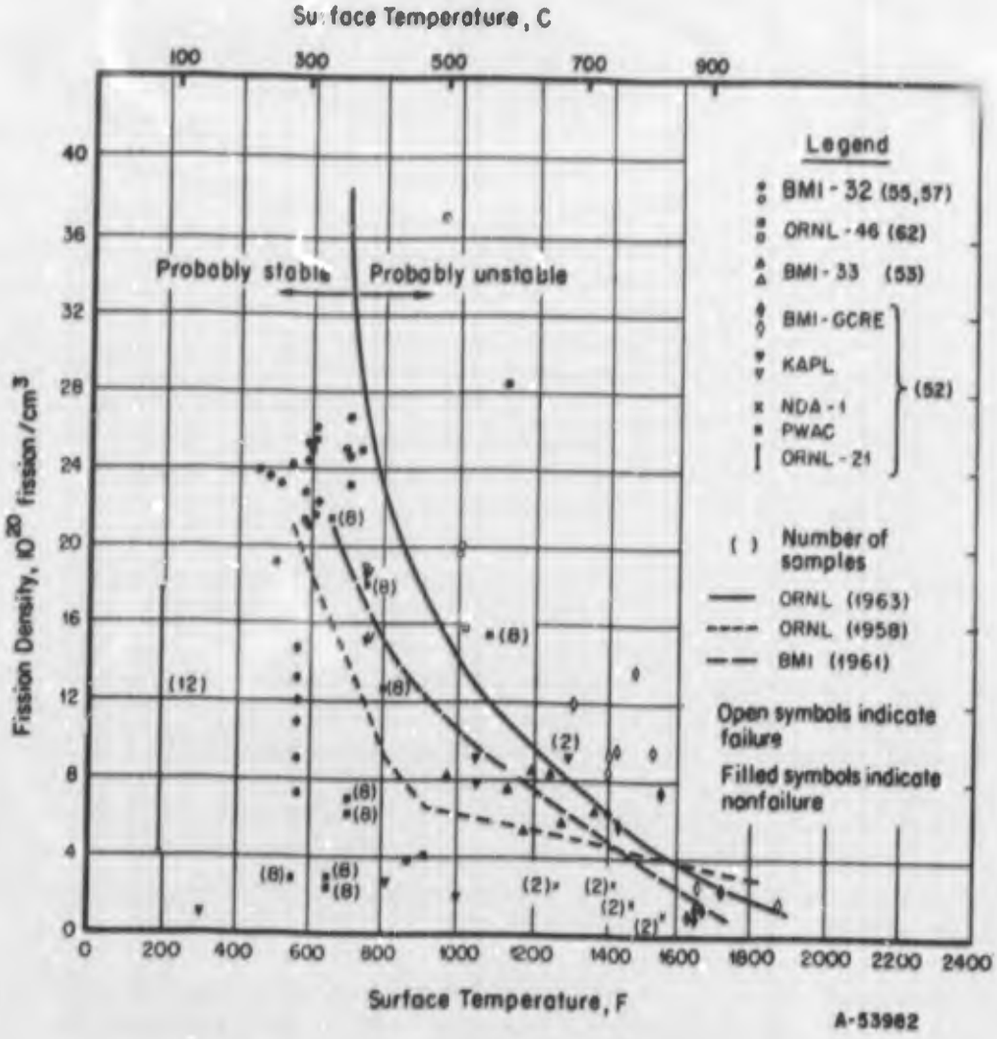


FIGURE 15. IRRADIATION PERFORMANCE OF STAINLESS STEEL-UO_2 DISPERSIONS[62]

Six stainless steel-24 weight percent UN and six stainless steel-24 weight percent UC dispersions were irradiated.(64-66) These specimens were irradiated at maximum center temperatures of 795 to 980 C and burnups up to 5.2×10^{20} fissions/cm^3 of fuel. Of the twelve specimens irradiated, four UC and one UN dispersion developed blisters. A minor reaction between UC and stainless steel was detected in a few of the specimens that had operated at the maximum temperatures. No reaction was found between the UN and the stainless steel, although a considerable amount of fission-gas-bubble formation had taken place. From this study, it appears that performance of UC and UN dispersions is comparable to that of UO_2 dispersions in stainless steel, provided the temperature is low enough to prevent a reaction between UC and the stainless steel matrix. The temperature at which reaction between UC and stainless steel takes place is about 870 C.

A stainless steel-30 weight percent UO_2 dispersion was irradiated together with three stainless steel-30 weight percent UN dispersions.(87) The fuel contained excess oxygen, thereby being UO_{2+x} rather than UO_2. The specimens were fabricated by sintering the roll bonding to a thickness of 0.045 inch. Irradiation temperatures of the specimens were in the 795 to 915 C range, and the burnup was 2×10^{20} fissions/cm^3 of fuel. No uniform thickness increases took place, but some blistering occurred on all specimens. The density decrease of the specimens was 0.6 to 1.5 percent, the UO_2 and UN dispersions performing about equally. Metallographic examination of the UO_2 dispersions showed that some reaction had taken place between the stainless steel and the excess oxygen from UO_2. It was also noted that the porosity of the UN particles had increased.

Two stainless steel-38 weight percent UN dispersions were irradiated, along with stainless steel-UO_2 dispersions.(67) These specimens operated at an estimated temperature of 500 to 700 C and burnups of 35×10^{20} fissions/cm^3 of fuel. One of the specimens failed and the other underwent considerable dimensional changes. The poor irradiation performance is believed to be due to the high temperatures during irradiation.

Irradiation testing has been performed with stainless steel-$(PuU)O_2$ dispersions by the British.(63) Two types of pins were tested, one type being sintered, the other hot swaged. The fuel volume in these dispersions varied from 30 to 50 volume percent. The hot-swaged pins were irradiated at center temperatures of 480 to 840 C with a maximum burnup of 8.8×10^{20} fissions/cm^3 of fuel. No dimensional changes took place in the pins that were irradiated below 700 C, but these specimens released up to 6.8 percent of their fission gases. Specimens irradiated above 700 C underwent some dimensional changes and released up to 100 percent of their fission gases. Metallographic examination shows the presence of fission-gas bubbles in the oxides and the healing of fabrication cracks in the matrix. The sintered dispersion pins were irradiated at surface temperatures of 585 to 645 C, the maximum burnup being 8.6×10^{20} fissions/cm^3 of fuel. Negligible dimensional changes occurred in the specimens, and the fission-gas release varied from 2.4 to 8.6 percent.

Irradiations of stainless steel-20 weight percent PuO_2 dispersions have been performed by Battelle-Northwest.(68,69) Tests were conducted on cold-pressed (82 percent theoretical density) and extruded (95 to 97 percent theoretical density) specimens. These specimens were irradiated to a burnup of about 8×10^{20} fissions/cm^3. Postirradiation annealing of the low-density specimens at 900 C for 1000 hours resulted in a density decrease of 1 percent. Annealing of the high-density specimens for 409 hours at 900 C resulted in a density decrease of 2.2 percent. It was concluded that the swelling was accommodated by the larger amount of porosity in the low-density specimens.

Nickel-Chromium Alloys

Considerable development work on UO_2 dispersed in various matrices was carried out by General Electric workers in their Nuclear Aircraft Propulsion program.[70] The proposed matrices and claddings were progressively changed as the temperature requirements of fuel specimens were increased. The following combinations of materials were considered:

Fuel Core	Cladding Material
310 stainless steel-UO_2	310 stainless steel
Nickel-chromium-UO_2	Nickel-chromium alloy
Niobium-UO_2	Iron-chromium-aluminum alloy
Chromium-titanium-UO_2	Iron-chromium, yttrium alloy
Iron-chromium-yttrium-UO_2	Iron-chromium-yttrium alloy
Chromium-titanium-UO_2	Chromium-yttrium alloy
Niobium-UO_2	Oxidation resistant coatings

Of the above systems, irradiation testing was carried out only on UO_2 dispersed in nickel-chromium and UO_2 dispersed in chromium-titanium. The other dispersions were not tested either because of incompatibility or because the temperature requirements were increased before irradiation of specimens was carried out.

Considerable numbers of irradiations were performed on UO_2 dispersed in a nickel-chromium alloy, with the specimens being also clad by a nickel-chromium alloy. Specimens having the following configurations were tested: Ribbons, round wire, concentric rings, corrugated plates, corrugated concentric rings, radial vane, lenticular wire, and wire screen. The tested specimens contained 30 to 60 weight percent UO_2, and were tested at temperatures of 980 to 1205 C. Testing times were up to 500 hours, the burnup being rather low. Most of these tests indicated no fission-gas release or dimensional changes. However, warping of the thin fuel elements was quite common.

Molybdenum-Uranium

A study was undertaken by APDA on UO_2 dispersions utilizing a molybdenum-uranium alloy as a matrix material.[71] The UO_2 content in the dispersion was 30 weight percent, the particles being less than 53 μ in size. These unclad plates were irradiated at surface temperatures of 315 to 370 C to a maximum burnup of 9.9×10^{20} fissions/cm^3 of fuel. Postirradiation examination showed some dimensional and density changes, the maximum density decrease being 3.7 percent (Table 8). Metallographic examination showed that neither the UO_2 nor the molybdenum-natural uranium matrix had changed as a result of irradiation. The dispersions were found to undergo less dimensional change than molybdenum-10 weight percent uranium alloys that had been exposed to the same irradiation conditions.

TABLE 8. IRRADIATION PERFORMANCE OF UO_2 DISPERSED IN A MOLYBDENUM-10 WEIGHT PERCENT URANIUM MATRIX

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density, % T.D.[a]	Specimen Shape	Width and Thickness, in.	Length, in.	Particle Size, μ	Cladding Material	Temperature, C		Irrad. Time, hr	Burnup, 10^{20} fissions/cc	Bulk Heat Gen. Rate, kw/cc	Surface Heat Gen. Rate, w/cm^2	Property Changes Caused by Irradiation, %				Reference
											Surface	Center					Thickness	Density	Volume	Length	
BMI-10-1	1	30	24	Mo-10 U	89-93	Plate	0.375 x 0.040	1.0	<53	Unclad	370	430	1800	4.3	2.1	87	1.5	-1.5	1.1	-.01	71
BMI-10-1	1	30	24	Mo-10 U	89-93	Plate	0.375 x 0.040	1.0	<53	Unclad	345	400	1800	4.2	2.0	83	2.2		0.53	Broken	71
BMI-10-2	1	30	24	Mo-10 U	89-93	Plate	0.375 x 0.040	1.0	<53	Unclad	370	400	4250	9.9	2.0	83	2.2	-2.9	1.2		71
BMI-10-2	1	30	24	Mo-10 U	89-93	Plate	0.375 x 0.040	1.0	<53	Unclad	315	345	4250	8.3	1.7	70	2.0	-3.7	1.6		71

(a) T.D. indicates theoretical density.

Refractory Metals

Because of demands for high specific power from fuel elements, it has become necessary to increase the fuel operating temperatures. Because of these high temperatures it is necessary to use refractory metals like niobium, molybdenum, and tungsten for matrix materials. As the temperature is increased, compatibility problems between the fuel material and the matrix become critical because of higher diffusion rates.

Cermets containing 80 volume percent UO_2 dispersed in niobium, chromium, and molybdenum were irradiated and examined by Battelle Memorial Institute.[72] The UO_2 particles in these specimens varied from 75 to 150 μ, and each particle was completely surrounded by the matrix. Because of accidental overheating in the capsules, all specimens were lost in one capsule, and partial melting occurred in one of the niobium-UO_2 cermets in the other capsule. All of the chromium-UO_2 and molybdenum-UO_2 dispersions were lost because of melting. The unmelted niobium-UO_2 cermet showed excellent dimensional stability, undergoing a density decrease of only 1.4 percent at a centerline temperature of 1405 C and a burnup of 7.9×10^{20} fissions/cm^3 of fuel. Both of the molybdenum-UO_2 specimens had cracked claddings and exhibited a central void caused by apparent fuel melting. The irradiation performances of the various fuel materials dispersed in refractory metal matrices are tabulated in Table 9.

Irradiation testing of tungsten-matrix cermet fuels has been carried out at Battelle-Northwest.[73,74] The specimens had a diameter of 0.200 inch and contained UO_2 particles of 75 to 250 μ hot pressed to 96 percent of theoretical density. The cermets were clad with 25 mils of tungsten. Short 2 to 4-hour irradiations showed that these tungsten-50 weight percent UO_2 cermets were chemically and dimensionally stable when irradiated at temperatures around 2100 C.

Longer irradiations were carried out on tungsten and molybdenum cermets containing UO_2 and UN.[75] Specimens were irradiated for 505 and 2,540 hours, and received burnups of 0.2 and 1.2×10^{20} fissions/cm^3, the irradiation temperature being 2000 and 2400 C. Fission-gas release from the specimens was quite high, ranging from from 15 to 67 percent. No dimensional or metallurgical changes occurred in the specimens as a result of irradiation.

A cermet containing niobium-30 weight percent UO_2 was irradiated to 0.34×10^{20} fissions/cm^3.[76] The maximum fuel center temperature was estimated to be 1500 C, and the heat-generation rate from the fuel was 905 w/cm^2. Ten percent of the fission gases were released from the fuel.

TABLE 9. IRRADIATION PERFORMANCE OF VARIOUS NUCLEAR FUELS DISPERSED IN REFRACTORY METAL MATRICES

Sample Designation	Number of Specimens in Sample	Fuel Weight, %	Fuel Volume, %	Matrix Material	Density % T.D.[a]	Specimen Shape	Diam., in.	Length, in.	Particle Size, μ	Cladding Material	Cladding Thickness, in.	Temperature, C		Irrad. Time, hr	Burnup, 10^{20} fissions/cm^3	Bulk Heat Gen. Rate, kw/cm^3	Surface Heat Gen. Rate, w/cm^2	Fission-Gas Release, %	Property Changes Caused by Irradiation, %		Reference
												Surface	Center						Diameter	Length	
BMI-39-1	1	84	80	Nb		Rod	.243	.975	75-150	Nb	.035	995	1590	1200					Specimen melted		72
BMI-39-1	1	84	80	Nb		Rod	.246	.975	75-150	Nb	.035	995	1590	1200					Specimen melted		72
BMI-39-2	1	84	80	Nb		Rod	.246	.975	75-150	Nb	.035	915	1405	1970	4.5-6.7				Specimen melted		.2
BMI-39-2	1	84	80	Nb		Rod	.246	.975	75-150	Nb	.035	915	1405	1970	7.9				0.7	0.2	72
BMI-39-1	1	81	80	Mo		Rod	.400	1.25	75-150	430 SS	.035	1195	1970	1200					Specimen melted		72
BMI-39-1	1	81	80	Mo		Rod	.400	1.25	75-150	430 SS	.035	1195	1970	1200					Specimen melted		72
BMI-39-2	1	81	80	Mo		Rod	.400	1.25	75-150	430 SS	.035	1010	1635	1970	4.5-6.7				5.8	0.3	72
BMI-39-2	1	81	80	Mo		Rod	.400	1.25	75-150	430 SS	.035	1010	1635	1970	4.5-6.7				4.8	1.3	72
GEH-14-416	1			Mo	98	Rod	.209	3.0		Mo		2330	2400	505	0.2	0.36	48	12			74-76
GEH-14-417	1			Mo	98	Rod	.209	3.0		Mo		2330	2400	2540	1.2	0.36	48				74-76
GEH-14-418	1			Mo	97	Rod	.209	3.0		Mo		2330	2370	505	0.2	0.34	45	12			74-76
GEH-14-419	1			Mo	97	Rod	.209	3.0		Mo		2300	2370	2540	1.2	0.34	45				74-76
GEH-14-414	1			W	96	Rod	.189	3.0		W	.025	2200	2370	505	0.2	0.34	42	22			74-76
GEH-14-415	1			W	96	Rod	.189	3.0		W	.025	2380	2450	505	0.2	0.45	54	61			74-76

(a) T.D. indicates theoretical density.

REFERENCES

(1) Seitz, F. S., and Koehler, J. S., "Displacement of Atoms During Irradiation", Solid State Phys., 2, 305 (1956).

(2) Kernohan, R. H., "Effect of Fissionable Particle Size on Fission Damage in Graphite" (July 14, 1954), ORNL-1722.

(3) Gilbreath, J. R., and Simpson, O. C., "The Effect of Reactor Irradiation on the Physical Properties of Beryllium Oxide", Proceedings of the Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, P/621.

(4) Yeniscavich, W., and Bleiberg, M. L., "The Effects of Irradiation on BeO-UO_2, ThO_2-UO_2 and ZrO_2-CaO-UO_2", Bettis Technical Review Reactor Technology (1960), WAPD-BT-20.

(5) Bleiberg, M. L., Berman, R. M., and Lustman, B., "Effects of High Burnup on Oxide Ceramic Fuels" (March, 1962), WAPD-T-1455.

(6) Brunhouse, J. S., et al., "Army Gas Cooled Reactor Systems Program In-Pile Test of Prototype ML-1 Fuel Elements" (June, 1964), IDO-28616.

(7) Vivian, G. L., "A Summary Report on ML-1 Fuel Element Development In-Pile Test History" (June, 1964), IDO-10040.

(8) Titus, G. W., and Saling, J. H., "Army Gas Cooled Reactor Systems Program High Temperature Irradiation of UO_2-BeO Bodies" (January, 1963), IDO-28600.

(9) Brunhouse, J. S., and Titus, G. W., "Evaluation of ML-1 Second Core Fuel Element Materials After Long Term In-Reactor Testing", Trans. ANS, 8, 43-44 (June, 1965).

(10) Saling, J. H., Wright, T. R., and Fromm, E. O., "A Study of the Radiation Stability of UO_2 in ZrO_2 in Comparison to UO_2 and UO_2 in BeO" (BMI Report to be published).

(11) Simnad, M. T., "Irradiation Effects on Dispersion Type BeO-UO_2 Fuels for EBOR" (October 7, 1963), GA-4643 (Part 2).

(12) Bennett, E. C., "Irradiation Testing of MGCR-HDR-1, -2, and -3 in the DR-1 Gas Loop", Topical Report (1961), HW-69403.

(13) Freas, D. G., et al., "A Study of the Radiation Stability of UO_2 Dispersions in BeO" (December 9, 1960), BMI-1484.

(14) Johnson, D. E., and Lofftus, F. H., "The Fabrication and Irradiation Testing of Dispersion-Type BeO-UO_2 Ceramic Fuel Materials" (October 10, 1962), GA-3483.

(15) Johnson, D. E., "Stability of BeO-UO_2 Reactor Fuel Material During Irradiation" (March 31, 1961), GA-2065.

(16) Johnson, D. E., et al., "Irradiation of BeO-UO_2 Ceramics" (1961), GA-2268.

(17) Johnson, D. E., and Mills, R. G., "Irradiation Behavior of BeO-UO_2 Fuel as a Function of Fuel Particle Size" (April 16, 1963), GA-4138.

(18) Rosenberg, H. S., Sunderman, D. N., and Diethorn, W. S., "The Study of Fission Gas Release from Neutron Activated Graphite and BeO Matrix Fuel Materials", Nuclear Reactor Chemistry. First Conference, Gatlinburg, Tennessee, October 12-14, 1960, TID-7610.

(19) Dickerson, R. F., and Dayton, R. W., "Progress Relating to Civilian Application During April 1964", BMI-1669.

(20) Dickerson, R. F., and Dayton, R. W., "Progress Relating to Civilian Application During June 1964", BMI-1674.

(21) Morgan, J. G., and Osborne, M. F., "Radiation Effects on Fueled BeO" (October 4, 1962), ORNL-TM-311.

(22) Manly, W. D., "Gas Cooled Reactor Program Quarterly Progress Report for Period Ending September 30, 1963", ORNL-3523.

(23) Manly, W. D., "Gas Cooled Reactor Program Quarterly Progress Report for Period Ending March 31, 1962", ORNL-3302.

(24) Hanna, G. L., Hickman, B. S., and Hilditch, R. J., "The Irradiation Behavior of Beryllium Oxide Dispersion Fuels" (March, 1963), AAEC/E-106.

(25) Hanna, G. L., "Further Examination of Irradiated Hot Pressed Beryllium Oxide Based Fuel Specimens" (August, 1964), AAEC/E-125.

(26) Hanna, G. L., and Hilditch, R. J., "The Irradiation Behavior of Cold Pressed and Sintered Beryllium Oxide Dispersion Fuels" (June, 1964), AAEC/E-124.

(27) Johnson, D. E., and Tobin, J. M., "The Dimensional Stability on Irradiation of Al_2O_3-UO_2, BeO-UO_2, Graphite-UC and Graphite-UC_2", Nuclear Reactor Chemistry. First Conference, Gatlinburg, Tennessee, October 12-14, 1960, TID-7610.

(28) Dayton, R. W., and Tipton, C. R., Jr., "Progress Relating to Civilian Applications During September 1960", BMI-1469.

(29) Stubbs, F. J., "Diffusion Studies in Dispersion-Type Fuels", Trans. Brit. Ceram. Soc., 62, 267-268 (1962).

(30) Durane, R. E., Klein, D. J., and Nykiel, F. R., "Effect of Reactor Irradiation on the Thermal Conductivity of Uranium Impregnated Graphite at Elevated Temperatures" (August 15, 1954), NAA-SR-836.

(31) Hunter, L. P., "Effect of Fission Recoil Fragments on the Thermal Conductivity of Graphite", J. Appl. Phys., 30 (12) (December 1959).

(32) The Pebble Bed Reactor Program. Summary of the First Capsule Irradiation (1959), AECU-4055.

(33) Morgan, J. C., and Osborne, M. F., "Irradiation Effects on UC_2 Dispersed in Graphite (ORNL-MTR-48-1), Interim Report No. 1 (August 18, 1960), CF-60-6-78.

(34) Gas Cooled Reactor Program Quarterly Progress Report for Period Ending June 30, 1961, ORNL-3166 (August, 1961).

(35) Davidson, J. M., and Helm, J. W., "The H-1 High Temperature Graphite Irradiation Experiment" (April, 1961), HW-64268.

(36) Davidson, J. M., and Helm, J. W., "The H-3 Irradiation Experiment: Irradiation of ESCR Graphite". Interim Report No. 1 (October 16, 1961), HW-71500A.

(37) Dayton, R. W., and Tipton, C. R., Jr., "Progress Relating to Civilian Applications During July 1959", BMI-1366.

(38) Goeddel, W. V., "The Development and Evaluation of Graphite-Matrix Fuel Compacts for the HTGR" (August 8, 1961), GA-2289.

(39) "40MW(E) Prototype High-Temperature Gas-Cooled Reactor Research and Development Program", Quarterly Progress Report for Period Ending September 30, 1960, GA-1774.

(40) Anderson, E. E., "Steady-State Release Function of Krypton and Xenon Fission Products at High Temperatures from (U, Th)C_2-Graphite Fuel Matrix in Out-of-Pile Experiments" (June 15, 1962), GA-3211.

(41) Hanna, G. L., Hickman, B. S., and Hilditch, R. J., "The Irradiation Behavior of Beryllium Based Dispersion Fuels – A Preliminary Irradiation Experiment" (1962), AAEC/E-96.

(42) Hanna, G. L., Hickman, B. S., and Hilditch, R. J., "The Irradiation Behavior of Hot-Pressed Dispersions of (U, Th)Be_{13} in Beryllium. Second Irradiation Experiment" (November, 1963), AAEC/E-112.

(43) Gibson, G. W., and Shupe, O. K., "Annual Progress Report on Fuel Element Development for FY 1961" (March 9, 1962), IDO-16727.

(44) Gibson, G. W., and Francis, W. C., "Annual Progress Report on Fuel Element Development for FY 1962" (August 15, 1962), IDO-16799.

(45) Graber, M. J., et al., "Results of ATR Sample Fuel Plate Irradiation Experiment" (March, 1964), IDO-16958.

(46) Huffman, J. R., "Materials Testing Reactor-Engineering Test Reactor Technical Branches Quarterly Report", October 1 to December 31, 1962, IDO-16857.

(47) Huffman, J. R., et al., "Materials Testing Reactor-Engineering Test Reactor Technical Branches Quarterly Report", January 1 to March 31, 1962, IDO-16781.

(48) Reinke, C. F., "Irradiation and Postirradiation Annealing of Some Aluminum-Base Fuels" (September, 1963), ANL-6665.

(49) Burian, R. J., Gates, J. E., and Dickerson, R. F., "Radiation Stability of Compacts of Aluminum and Fuel-Bearing Glass Fibers" (March 30, 1962), BMI-1573.

(50) Lambert, J. D. B., "Irradiation Stability of Dispersion Fuels", Trans. Brit. Ceram. Soc., 62, 247-256 (1963).

(51) Barney, W. K., Ray, W. E., and Sowman, H. C., "Development of Ferritic Stainless Steel-UO_2 Dispersion Fuel Elements" (January 30, 1958), KAPL-1908.

(52) Haynes, V. O., Neill, F. H., and Schaffner, L. D., "Summary of UO_2-Stainless Steel Dispersion Element Irradiation Experiment" (March 18, 1958), CF-58-2-71.

(53) Freas, D. G., et al., "Radiation Stability of Spherical and Hydrothermal UO_2 Dispersed in Type 318 Stainless" (July 14, 1961), BMI-1529.

(54) Nash, R. R., "Nuclear Fuels Newsletter" (February 1959), WASH-705.

(55) Paprocki, S. J., et al., "Fabrication and Irradiation of SM-2 Core Materials" (July 12, 1964), BMI-1528.

(56) Hamilton, D. B., et al., "Irradiation-Capsule Study for SM-2 Reference and Alternate Dispersion Fuels" (May 25, 1961), BMI-1516.

(57) Schaffner, L. D., "Army Reactor Program Annual Progress Report for Period Ending October 31, 1962" (April 10, 1963), ORNL-3386.

(58) Bates, J. L., "Metallic Uranium in Irradiated UO_2", HW-82263.

(59) Daniel, R. C., et al., "Effects of High Burnup on Zircaloy-Clad, Bulk UO_2, Plate Fuel Element Samples" (September, 1962), WAPD-263.

(60) Schaffner, L. D., "Army Reactor Program Progress Report" (February 14, 1962), ORNL-3231.

(61) Schaffner, L. D., "Army Package Power Reactor Project Annual Progress Report for Period Ending January 31, 1960" (April 4, 1960), ORNL-2907.

(62) Thurber, W. C., et al., "Irradiation Testing of Fuel for Core B of the Enrico Fermi Fast Breeder Reactor" (November, 1964), ORNL-3709.

(63) Frost, B. R. T., et al., "Fabrication and Irradiation Studies of UO_2-Stainless Steel and $(U, Pu)O_2$-Stainless Steel Cermets", Third International UN Conference on the Peaceful Uses of Atomic Energy (May 1964), A/CONF-28/P/153.

(64) Dayton, R. W., and Tipton, C. R., Jr., "Progress Relating to Civilian Applications During December 1958", BMI-1958.

(65) Dayton, R. W., and Tipton, C. R., Jr., "Progress Relating to Civilian Applications During March 1959", BMI-1330.

(66) Dayton, R. W., and Tipton, C. R., Jr., "Progress Relating to Civilian Applications During June 1959", BMI-1357.

(67) Gates, J. E., et al., "Experiments to Determine the Radiation Stability of UN Dispersions in Stainless Steel" (June 14, 1960), BMI-1446.

(68) Reactor and Materials Technology, Monthly Report, March 1965, BNWC-83A.

(69) Reactor and Materials Technology, Monthly Report, June 1965, BNWC-86A.

(70) Lever, R. C., Malon, C. P., and Holteiman, M. L., "Metallic Fuel Element Materials" (June 20, 1962), APEX-913.

(71) Keller, D. L., et al., "Fabrication Development and Irradiation of Uranium-Molybdenum Alloy-UO_2 Dispersion" (March 12, 1959), BMI-APDA-647.

(72) Keller, D. L., et al., "High Temperature Irradiation Test of UO_2 Cermet Fuels" (January 7, 1963), BMI-1608.

(73) Hendrickson, N. M., and Green, J. K., Quarterly Progress Report. Research and Development Programs Executed for the Division of Reactor Development, July, August, September 1963, HW-79280.

(74) Reactor and Fuels Research and Development Operation Monthly Report (October, 1962), HW-75376A.

(75) Ceramics Research and Development Operation Quarterly Report, October to December 1963, HW-76304.

(76) Reactor and Fuels Laboratory, Monthly Report (July, 1964), HW-83445A.

MK/spr:cm

APPENDIX

TABLE A-1. PHYSICAL PROPERTIES OF SELECTED FUEL AND MATRIX MATERIALS*

	Density, g/cm^3	Crystal Structure	Melting Point, C	Total Cross Section, Σ_a(cm^{-1})	Thermal Expansion, percent			Thermal Conductivity, cal/(sec)(cm)(C)			
					RT-500 C	RT-1000 C	RT-1500 C	RT	500 C	1000 C	1500 C
Matrix Materials											
BeO	3.0	HCP	2530	0.00073	0.28	0.90	1.6	0.53	0.15	0.05	0.04
Al_2O_3	3.9	HCP	2045	0.011	0.38	0.85	1.45	0.084	0.023	0.015	0.015
MgO	3.6	FCC	2947	0.0034	0.65	1.35	2.30	0.092	0.033	0.014	0.014
Graphite	2.2	HCP	3700(a)	0.00038	0.16(b)	0.40(b)	0.55(b)	0.38	0.23	0.14	0.12
					0.09(c)	0.25(c)	0.38(c)				
Beryllium	1.8	HCP	1277	0.0012	0.8	2.0	(d)	0.41	0.25	0.17	(d)
Aluminum	2.7	FCC	660	0.014	1.3	(d)	(d)	0.54	0.45	(d)	(d)
430 Stainless Steel	7.7	BCC	1400	0.241	0.32	0.73	(d)	0.060	0.063	0.074	(d)
347 Stainless Steel	8.0	FCC	1400	0.261	0.52	1.12	(d)	0.036	0.053	0.068	(d)
Niobium	8.6	BCC	2468	0.063	0.40	0.88	1.3	0.11	0.13	0.14	0.16
Molybdenum	10.2	BCC	2610	0.173	0.28	0.60	0.95	0.33	0.29	0.24	0.20
Tungsten	19.3	BCC	3410	1.19	0.20	0.47	0.75	0.38	0.31	0.26	0.25
Fuel Materials											
UO_2	10.9	FCC	2760	16.63(e)	0.50	1.05	1.8	0.09	0.04	0.03	0.03
UN	14.3	FCC	2800	23.70(e)	0.40	0.91	1.5	0.32	0.45		
UC	13.6	FCC	2450	22.68(e)	0.51	1.1	1.7	0.09	0.06	0.05	
ThO_2-5 w/o UO_2	10.1	FCC	3240	0.91(e)	0.43	0.95	1.5	0.036	0.013	0.008	0.005
PuO_2	11.5	FCC	2280	28.8	0.45	1.12					

(a) Sublimation temperature.
(b) Perpendicular to extrusion.
(c) Parallel to extrusion.
(d) Molten.
(e) Fully enriched uranium (93 percent).
*Sources:

(1) Goldsmith, A., Waterman, T. E., and Hirschhorn, H. J., Handbook of Thermophysical Properties of Solid Materials, The Macmillan Co., New York (1961).
(2) Tietz, T. E., and Wilson, J. W., Behavior and Properties of Refractory Metals, Stanford University Press (1965).
(3) Endebrock, R. W., "Properties of Fuels for High Temperature Reactor Concepts" (November 1, 1962), BMI-1598.
(4) Burdi, G. F., SNAP Technology Handbook. Refractory Fuels and Claddings, Vol III (April 30, 1965), NAA-SR-8617, Vol III.

TABLE A-2. COMPATIBILITY BETWEEN SELECTED FUEL AND MATRIX MATERIALS*

	Temperature at Which Reaction Begins, C				
	UO_2	UC	UN	PuO_2	ThO_2-UO_2(a)
BeO	1800(1)				1500(5)
Al_2O_3	1800(1)			1590(2)	
MgO	1800(1)			1590(2)	
Graphite	1300(2)	1600(2)			
Beryllium	600(1)	700(2)			260(6)
Aluminum	500(1)(b)	600(2)	400(4)		260(6)
430 Stainless Steel	1315(2)	870(1)	1300(4)	1400(2)	1095(6)
447 Stainless Steel	1315(2)	870(1)	1300(4)	1400(2)	1095(6)
Niobium	1400(1)	1100(1)		1400(2)	815(6)
Molybdenum	2600(3)	1200(1)		1370(2)	
Tungsten	2760(3)	1800(2)		1370(2)	

(a) By thermodynamic considerations, ThO_2-UO_2 should be more compatible than UO_2 with all materials.

(b) Indications are that irradiation enhances reactions between UO_2 and aluminum, and thereby lowers the compatibility temperature.

*Sources:

(1) Burdi, G. F., "SNAP Technology Handbook. Refractory Fuels and Claddings", Vol III (April 30, 1965), NAA-SR-8617, Vol III.

(2) Endebrock, R. W., "Properties of Fuels for High Temperature Reactor Concepts" (November 1, 1962), BMI-1598.

(3) Seddon, B. J., "Summarized Compatibility Review of Reactor Materials for Co_2-Cooled Graphite Moderated Reactors", TRG-Report-842(R) (1965).

(4) Price, D., et al., "Compatibility of UN With Potential Cladding Metals", BMI-1760 (February 8, 1966).

(5) Hanna, G. L., and Hilditch, R. J., "The Irradiation Behavior of Cold Pressed and Sintered Beryllium Oxide Dispersion Fuels" (June, 1964), AAEC/E-124.

(6) Paprocki, S. J., and Dickerson, R. F., "Properties of Reactor Materials", Nucleonics, Reference Data Manual (1960).

TABLE A-3. ANNEALING TEMPERATURE FOR REMOVING FAST-NEUTRON DAMAGE IN SELECTED MATRIX MATERIALS

Material	Temperature, C	Source
BeO	1000-1500(a)	1
Al_2O_3	500-700(a)	1
Graphite	1000-1200(a)	1
Aluminum	140	2
Stainless Steel	500	2, 3
Niobium	490	2
Molybdenum	800	4
Tungsten	1090	4

(a) The required annealing temperature seems to differ with batch of material. Also, materials of a higher density require a higher annealing temperature. It was found also that irradiation of a material at a temperature is considerably more effective than post-irradiation anneal at the same temperature.

*Sources:

(1) Kircher, J. F., and Bowman, R. E., Effects of Radiation on Materials and Components, Reinhold Publishing Company, New York (1964).

(2) Aqua, E. N., and Allio, R. J., "Radiation Effects Upon and Recovery of the Mechanical Properties of Metals", KAPL-2103.

(3) Busy, S. H., Irradiation Effects in Cladding and Structural Materials, ASTM, Rowman and Littlefield, Inc., New York (1965).

(4) Quarterly Progress Report: "Irradiation Effects on Reactor Structural Materials, February, March, April, 1965", BNWL-94.

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6 (First Addendum)	A Survey of Current Research and Developments in the Field of Dosimetry (March 31, 1959), AD 210766
*19	The Effect of Nuclear Radiation on Lubricants and Hydraulic Fluids (May 31, 1961), AD 261278 (Supersedes Report No. 4)
20	The Effect of Nuclear Radiation on Structural Metals (September 15, 1961), AD 265839 (Supersedes Report No. 5) AD 265839
21	The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials (September 1, 1961), AD 267890 (Supersedes Reports Nos. 3, 9, 13, and 17 and Memos Nos. 1, 3, 8, 13, and 17)
21 Addendum	The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials (August 31, 1964), AD 454056 (Supersedes Reports Nos. 3, 9, 13, and 17 and Memos Nos. 1, 3, 8, 13, and 17)
23	Proton and Electron Damage to Solar Cells (April 1, 1962), AD 274954
27	The Effect of Nuclear Radiation on Ceramic Reactor-Fuel Materials (June 30, 1963), AD 414669
29	Effect of Nuclear Radiation on Metallic Fuel Materials (September 30, 1963), AD 420282
30	Effect of Nuclear Radiation at Cryogenic Temperature on Semiconductor Materials (September 30, 1963), AD 418420
31	Part I. A Survey of Irradiation Facilities (September 30, 1963), AD 418579 Part II. A Survey of Particle Accelerators (September 30, 1964), AD 447307
32	Space Radiation Damage to Electronic Components and Materials (September 30, 1963), AD 422325
33	Survey of Current Research and Developments in the Field of Dosimetry (September 30, 1963), AD 419304

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35	Radiation Effects Experimental Procedures (August 31, 1964), AD 447977
36	Effects of Nuclear Radiation on Electronic Components Including Semiconductors (October 1, 1964), (Supersedes Report Nos. 1, 1A, 10, 10A, and 18), AD 454707
37	The Space Radiation Environment and Its Interaction With Matter (December 31, 1964) (Supersedes Memorandum No. 21), AD 456854
38	Radiation Effects State of the Art 1964-1965 (June 30, 1965), AD 473472
39	Space Radiation Damage to Electronic Components and Materials (January 31, 1966)

Classified Reports

Report Number	Title
*1-C	The Effect of Nuclear Radiation on Hydraulic, Pneumatic, and Mechanical Systems for Subsonic, Transonic, and Low-Supersonic Speed Aircraft (Title Unclassified) (Secret, Restricted Data) (May 31, 1958)
*1-C (First Addendum)	The Effect of Nuclear Radiation on Hydraulic, Pneumatic, and Mechanical Systems for Subsonic, Transonic, and Low-Supersonic Speed Aircraft (Title Unclassified) (Secret, Restricted Data) (March 31, 1959)
*1-C (Second Addendum)	The Effect of Nuclear Radiation on Hydraulic, Pneumatic, and Mechanical Systems (Title Unclassified) (Secret, Restricted Data) (September 15, 1960)
*2-C	The Effect of Nuclear Radiation on Ceramic Materials (Title Unclassified) (Secret) (June 30, 1959) AD 157173
*2-C (First Addendum)	The Effect of Nuclear Radiation on Ceramic Materials (Title Unclassified) (Secret, Restricted Data) (June 15, 1961)
*4-C	The Effect of Nuclear Radiation on Electrical and Electronic Systems (Title Unclassified) (Secret, Restricted Data) (March 15, 1960)
*6-C	The Effects of Nuclear Weapon Bursts and Simulated Bursts on Electronic Components (Title Unclassified) (Secret, Restricted Data) (May 31, 1961)

Unclassified Memoranda

Memorandum Number	Title
*9	The Effect of Nuclear Radiation on Glass (November 30, 1958), AD 207701
*10	Format for Reporting Radiation Effects Data (May 15, 1959), AD 218251
*11	The Effect of Nuclear Radiation on Hydrocarbon Fuels (November 30, 1958), AD 207702
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24	A Selected Bibliography of Pulsed Radiation Effects (June 1, 1963), AD 615339
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28	Radiation Effects on Propellants and Explosives: An Annotated Bibliography (October 29, 1965), AD 474256
29	Compilation of Nuclear, Pulsed, and Space Radiation Effects Projects, (April 15, 1966), AD 481296

Classified Memoranda

Memorandum Number	Title
1-C	The Effects of Nuclear Radiation on Fluorolubes and Other Gyroscope Fluids (Title Unclassified) (Secret) (September 5, 1958), AD 302126
2-C	The Effect of Nuclear Radiation on Explosives and Solid Propellants (Title Unclassified) (Secret, Restricted Data) (July 15, 1959), AD 342563
3-C	Dose-Rate Effects on Materials, Components, and Systems (Title Unclassified) (Secret, Restricted Data) (July 31, 1959), AD 341033

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13. ABSTRACT

A complete summary of the irradiation performance of the various fuel materials dispersed in different matrices has been collected. The irradiation performance of the following dispersions is discussed in the report: BeO-UO_2, BeO-UO_2-ThO_2, Al_2O_3-UO_2, MgO-UO_2, UO_2, UC, UC_2, and $(U,Th)C_2$ dispersed in graphite; UBe_{13} and $(U,Th)B_{13}$ dispersed in beryllium; UO_2, U_3O_8, U_3O_8 glass, and UAl_3 dispersed in aluminum; UO_2, UC, and UN dispersed in stainless steel; UO_2 and UO_2-Y_2O_3 dispersed in nickel-chromium alloys; UO_2 dispersed in niobium, UO_2 dispersed in molybdenum, UO_2, UN, and UC dispersed in tungsten. The data which is summarized includes percentage of fuel by weight and volume, specimen shape and size, fuel particle size, cladding material and clad thickness, irradiation temperature, heat generation rates, irradiation times, fuel burnup, fission gas release, dimensional and density changes resulting from irradiation.

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14. KEY WORDS	LINK A		LINK B		LINK C	
	ROLE	WT	ROLE	WT	ROLE	WT
Radiation Effects						
State-of-the Art						
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Uranium Oxide						
Thorium Oxide						
Aluminum Oxide						
Magnesium Oxide						
Uranium Carbide						
Graphite						
Uranium Beryllide						
Beryllium						
Aluminum						
Uranium Nitride						
Stainless Steel						
Yttrium Oxide						
Reactor Fuels						
Dispersion Fuels						
Fission Products						
Nickel Alloys						
Chromium Alloys						
Niobium						
Molybdenum						
Tungsten						