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RERTR-3 FUEL TEST SPECIMENS**

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ABSTRACT

The RERTR-3 irradiation test was designed to investigate the irradiation behavior of aluminum matrix U-Mo alloy dispersion fuels under high-temperature, high-fission-rate conditions. Initial postirradiation examination of RERTR-3 fuel specimens has concentrated on binary U-Mo atomized fuels. The rate of matrix aluminum depletion was found to be higher than predictions based on low temperature irradiation data. Wavelength Dispersive X-ray Spectroscopy (WDS) indicates that aluminum is present in the interior of the fuel particles. WDS data is supported by a mass and volume balance calculation performed on the basis of image analysis results. The depletion of matrix aluminum seems to have no detrimental effects on fuel performance under the conditions tested to date.

INTRODUCTION

The RERTR-3 irradiation test was inserted into the ATR (Advanced Test Reactor) during the last RERTR meeting, and began irradiation on October 7, 1999. [1] The RERTR-3 test was designed to investigate the critical issue of the fuel performance of U-Mo alloy dispersions during irradiation at high power and high temperature. A series of U-Mo alloys with compositions ranging from 6.7 to 10.6 wt.% molybdenum were irradiated to differentiate compositional effects. Atomized U-10Mo powder and U-10Mo filings were subjected to heat treatments prior to irradiation in order to determine the affects of microstructure on fuel performance.

The RERTR-3 experiment consisted of six flow-through capsules stacked vertically inside of an outer basket. Two columns of four fuel test specimens were arranged within each capsule, for a total of eight specimens per capsule. The experiment was irradiated for a total of 48 effective full-power days (EFPDs) over two reactor cycles. Peak plate burnup ranged between 24.9 at.% U-235 for the plate farthest above the core centerline up to 41.0% for plates near the core centerline. The desire to irradiate at high flux limited the choice of flow-through irradiation positions in the ATR. The B-7 position, a flow-through hole with a diameter of 22.2 mm outboard of driver fuel element 33 was used for this test. The small diameter of this position necessitated the use of small irradiation test specimens; each test coupon was 10.0 mm x 41.1 mm x 1.52 mm, and contained approximately 0.6 grams of fuel.

Because of the small size of these specimens, analysis of fuel behavior during irradiation depends heavily on metallography.

Postirradiation examination has so far concentrated on analysis of plates V03, V07, and S03. These three plates give an overview of the effects of temperature and composition on fuel performance. Plate V03 is a U-10.6Mo (wt.%) plate irradiated at high temperature; plate V07 is the same composition irradiated at lower temperature. Plate S03 is a U-6.7Mo plate irradiated at high temperature. In addition, a look at plate E01 shows the effect of a small osmium addition on fuel-matrix interaction under irradiation. Fission density, burnup, and temperature data for these fuel plates and two U-10.3Mo plates from the RERTR-1 and -2 experiment are given in Table 1.

Table 1. Fission density, burnup, and fuel centerline temperatures of irradiated fuel test coupons from RERTR-3 and RERTR-2

Plate ID	Nominal Fuel composition (wt.%)	Analyzed fuel composition (wt.%)	²³⁵ U burnup ² (peak %)	Fuel fission density (10 ²¹ cm ⁻³)	Avg. fuel fission rate (10 ¹⁴ cm ⁻³ s ⁻¹)	Fuel ³ Centerline Temperature (°C at BOL)
V03	U-10Mo	U-10.6Mo	38	2.6	6.3	196
V07	U-10Mo	U-10.6Mo	32	2.1	5.1	158
S03	U-6Mo	U-6.7Mo	39	2.9	7.0	204
E01	U-6Mo-1.7Os	U-7.0Mo-1.7Os	39	2.8	6.8	203
V002 ¹	U-10Mo	U-10.3Mo	38	2.7	3.3	65
V003 ¹	U-10Mo	U-10.3Mo	70	5.0	2.5	65

¹RERTR-1 and -2 experiments. ²Calculated burnup. ³Based on a three dimensional model.

Beginning of life (BOL) fuel centerline temperatures were calculated based on a three-dimensional model. These are used as the reference temperature for comparison of fuel plate behavior. Owing to the large amount of fuel-matrix interaction and the small plate size, the complete thermal history of the RERTR-3 plates has not yet been determined. A model is being developed to reconstruct the thermal history of the fuel plates as a function of power and change in thermal conductivity due to fission and interaction layer growth. Because of interaction layer growth, the peak fuel meat temperature may be higher than the BOL temperature.

MEASUREMENT METHOD

Test plates were sectioned at measured locations, mounted, and polished for metallographic examination. Measurements of the area fractions of aluminum, interaction product, and unreacted fuel were made on the basis of gray scale contrast. Two types of images were examined. Low magnification montages (75X) of the length and cross section of fuel zones were used to measure the average fraction of unreacted fuel in each plate, and to determine a confidence interval for the measurement. High magnification images of smaller regions were used to carefully measure the fraction of fuel, interaction, and unreacted aluminum. Since the plates made using spherical powder appear to have an isotropic distribution of fuel, area fraction translates directly to volume fraction. Examples of as-fabricated and irradiated U-10Mo fuel are shown in Figure 1.

Micrographs of the other fuel plates discussed here can be found elsewhere in these proceedings. [2]

Metallographic measurements and thickness measurements were combined to analyze the behavior of the fuel. Prior to irradiation, a thin boehmite ($\text{AlO}(\text{OH})$) layer was grown on the test plates by autoclaving in water at 180°C and saturation pressure (0.9 Mpa, 130 psi). This film served to protect the plates from the corrosion observed during

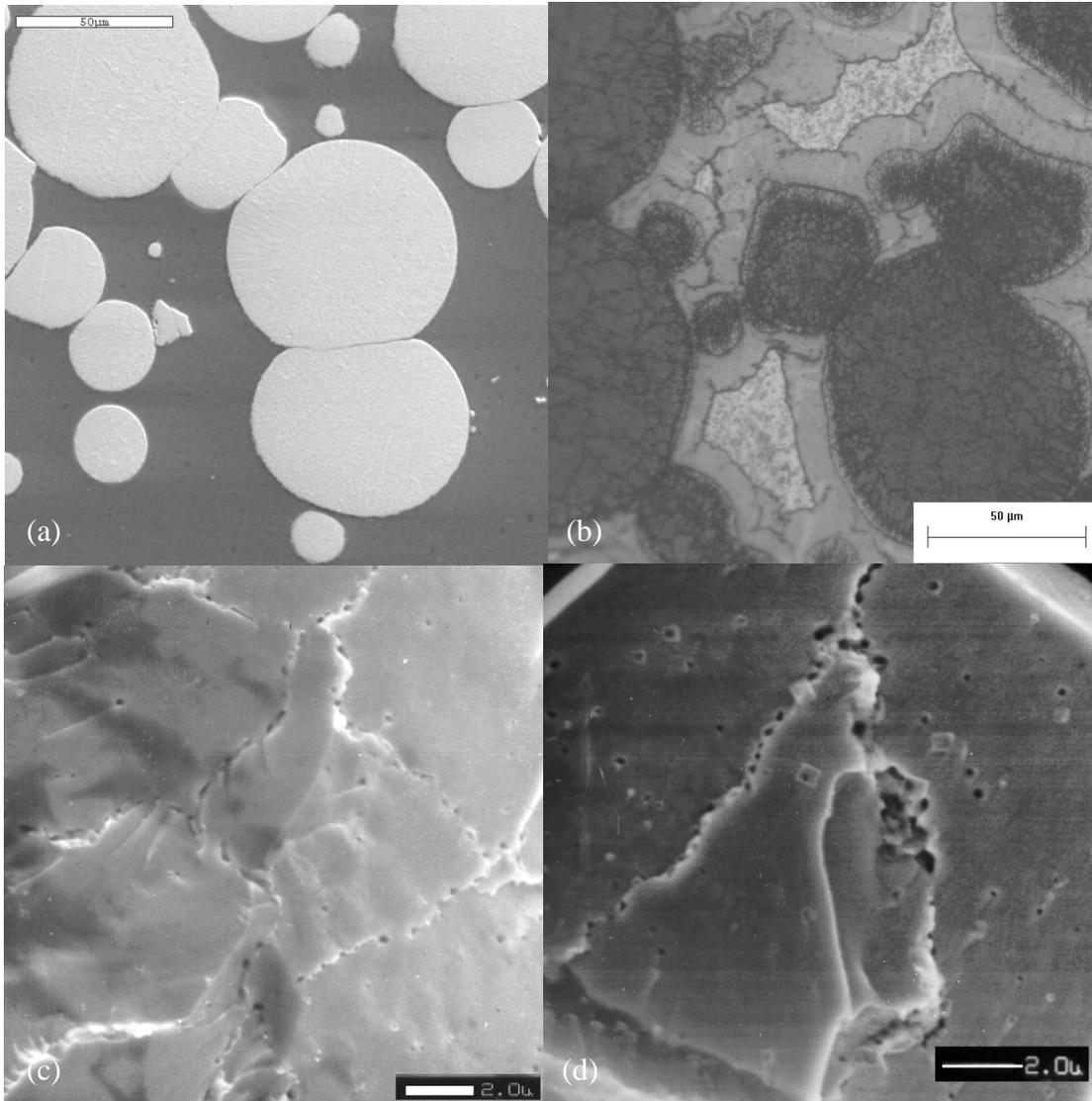


Figure 1. Microstructure of U-10Mo atomized fuel. (a) SEM micrograph of as-fabricated plate. (b) Etched optical microstructure of plate V03 after 37.6% U-235 burnup. Same magnification as (a). (c) SEM micrograph of V03 fracture surface. (d) SEM of RERTR-1 plate V002 after 38.4% burnup.

irradiation of the RERTR-1 and -2 experiments.^a A thin, uniform surface film 3-5 μm thick was observed on sectioning the irradiated specimens. A thickness measurement taken outside of the fuel zone was subtracted from the average of postirradiation thickness measurements inside of the fuel zone to determine an average plate thickness change. The entire thickness change was assigned to fuel meat swelling, and fractional swelling was calculated based on nominal as-fabricated meat thickness. The as-fabricated meat thickness of the RERTR-3 specimens was 0.76 mm (0.030 in.).

Owing to the small size of these plates, there were relatively large temperature gradients in both the lateral and transverse directions. This resulted in corresponding gradients in thickness change and fuel-matrix reaction rate. Average plate thickness change and average component volumes across all available metallographic sections were used for the analysis reported in this paper. The results of metallographic and thickness measurements are given in Table 2.

Table 2. Results of metallographic and dimensional measurement.

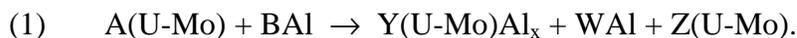
Plate ID	As-fabricated fuel vol. %	Postirradiation measurements				
		Fuel vol. %	Interaction vol. %	Aluminum vol. %	Thickness change (avg.) mm/(inch.)	Average meat swelling (%)
V03	51.9	59.5	23.5	17	0.109/(0.0043)	14.3
V07	51.9	61.4	11.5	27.1	0.015/(0.0006)	2.1
S03	50.5	56.7	35.0	8.3	0.104/(0.0041)	13.7
E01	51.1	46.2	48.6	5.2	0.145/(0.0057)	19.0
V003 ¹	28.3	26.1	11.0	62.9	0.038/(0.0015) ²	9.0

¹RERTR-2 specimen. ²This value may be skewed owing to surface corrosion/erosion during irradiation.

DATA ANALYSIS

Examination of Figure 1 and Table 2 shows that as a result of irradiation, a large amount of aluminum was consumed by reaction with fuel, and that the fuel volume fraction has increased. Examination of fuel fracture surfaces by SEM (Scanning Electron Microscopy) (fig. 1(c)) revealed a bubble distribution similar to that found in fuel at similar burnup from the RERTR-1 experiment (fig. 1(d)); there were no signs of gross fission-gas-driven swelling.

Conservation of volume and mass were used to fit the data of Table 2 to a chemical equation of the form:



Here, A, B, Y, W, and Z are molar quantities, and x determines the stoichiometry of the (U-Mo)Al_x interaction phase. Equation (1) assumes a solid solution of Mo in UAl_x. The

^a There were no plate failures due to pitting corrosion in RERTR-3; RERTR-1 and -2 did not receive the prefilm treatment and exhibited five instances of pitting corrosion.

quantities A and B are known from as-fabricated data, and W is deduced from postirradiation metallographic and meat swelling measurements. If it is assumed that the aluminum does not undergo a significant volume change on irradiation, the quantity $\Delta A = B - W$ determines the amount of aluminum consumed to form the $(U-Mo)Al_x$ interaction phase. The measured volume of this phase is then used to compute a required $(U-Mo)Al_x$ density. Assuming a value for x determines the values Y and Z. Z is used to calculate alloy density and fuel swelling. Solutions for $x = 1, 2, 3,$ and 4.44 are given in Table 3 for plate V03. Both the addition of Mo to UAl_x and irradiation induced swelling would be expected to reduce the density of $(U-Mo)Al_x$; however, the calculated density of the interaction product is larger than that of the corresponding pure UAl_x compound. The calculated U-Mo density decrease is also at odds with the gas bubble population and size distribution shown in Figure 1. Reference [1] shows that swelling in these fuel plate specimens is closely related to formation of the fuel-matrix reaction product. It is thus apparent that there is no reasonable solution to equation (1) using the values in Table 2.

Table 3. Possible solutions to equation (1) for U-10.6Mo plate V03 for $x=2-4.44$.

x in $(U-Mo)Al_x$	Y	Calculated $(U-Mo)Al_x$ density	Accepted UAl_x density	Z=A-Y	Calculated U-Mo density	Calculated U-Mo swelling (%)
2	1.30	12.7	8.14	3.04	9.3	45
3	0.86	9.2	6.8	3.48	10.6	38
4.44 ¹	0.59	7.2	5.7	3.75	11.4	33

¹The compound ' UAl_4 ' has a solubility range of 80-82.8 at.% Al, nominally $U_{0.9}Al_4$ or $UAl_{4.44}$.

The most plausible explanation, then, for the shortage of matrix aluminum and the volume increase in the fuel phase is the dissolution of a small amount of aluminum into the fuel alloy. That this is possible is confirmed by phase-equilibria studies. Based on data from three researchers, the maximum equilibrium solubility of aluminum in γ -U was assessed as 4.7 at.% at 1105°C. [3] One researcher found a maximum solubility of 6.9 at.%, presumed to be due to a nonequilibrium condition. It has been found that fission-related processes cause U-Mo alloys to be maintained in the γ -phase over a wide range of temperature and fission rates where it does not exist in equilibrium. [4] The solubility range for Al in U-Mo fuels under irradiation should thus be similar to that of γ -U.

The actual presence of aluminum in the interior region of fuel particles was confirmed by WDS measurements. Data presented in Table 4 are based on standardless analysis (ZAF corrected), due to the lack of a suitable standard at the time the examination was made. Analysis of two particles shows that the aluminum content in the particle interiors ranged from 1.4 to 4.6 wt.%. WDS data also indicate that the interaction phase contains all three of the major elements present, and may be of the form^b $(U-Mo)Al_x$. The composition is aluminum rich.

^b The interaction layer appears to be single phase from SEM and optical microscopy, but could also be a fine mixture of $MoAl_x$, UAl_x , and $(U-Mo)Al_x$ phases, undetectable using the examination methods used to date.

Table 4. WDS analysis results for U-8.2Mo plate R04.

Measurement ID	Wt.%			At.%			Mo ^c U+Mo	Stoichiometry of (U-Mo)Al _{x,q}
	Mo	Al	U	Mo	Al	U		
Interior-4	8.7	1.4	89.9	17.4	10.0	72.6	0.083	q = 0.11
Interior-5	9.3	4.6	86.1	15.4	27.1	57.6	0.092	q = 0.35
Interaction-4	5.6	45.4	49.0	3.0	86.4	10.6	0.097	x = 6.1
Interaction-5	5.6	46.0	48.4	3.0	86.7	10.4	0.098	x = 6.2

A more general equation for the fuel-matrix reaction that accounts for the possibility of incorporation of aluminum in U-Mo can be written as equation (2):



Here (U-Mo)Al_q denotes a solid solution of aluminum in γ -phase (U-Mo) alloy. A series of conservation of mass and volume equations can be written that allow a solution for Y, Z, x, and q. These include an equation for total fuel volume (eqn. 3), and equations for the volumes of the (U-Mo)Al_q (eqn. 4) and (U-Mo)Al_x (eqn. 5) phases.

$$(3) \quad V_m = V_{(\text{U-Mo})\text{Al}_q} + V_{(\text{U-Mo})\text{Al}_x} + V_{\text{Al}}$$

$$(4) \quad V_{(\text{U-Mo})\text{Al}_q} = ZG(q)(1+S_1)/J(q)$$

$$(5) \quad V_{(\text{U-Mo})\text{Al}_x} = YL(x)(1+S_2)/H(x)$$

Here V_m is the volume of the irradiated fuel meat relative to the original volume, S_1 and S_2 are swelling of the U-Mo-Al solid solution and (U-Mo)Al_x phases, respectively, and $G(q)$ and $L(x)$ are the molecular weights of the U-Mo-Al solid solution and intermetallic phases, respectively. $J(q)$ is the density of the U-Mo-Al phase, and $H(x)$ is the density of the (U-Mo)Al_x intermetallic phase. $J(q)$ is calculated by assuming a solid solution and using the rule of mixture based on the atom fractions of U, Mo, and Al present. A relationship for $H(x)$ was established by using linear regression to fit the known densities of the three UAl_x phases [5] as a function of x (eqn. 6).

$$(6) \quad H(x) = 9.38 - 0.98x$$

The value of the intercept was adjusted to account for the molybdenum content in

^c Because these data were taken without reference to an external standard, an assessment of reliability is made based on internal consistency. Preirradiation chemical analysis of the (nominally) U-7Mo fuel gave a composition of 8.2±0.8 wt.% molybdenum. When the analyzed uranium content of the irradiated fuel is adjusted for depletion to the preirradiation condition, and the molybdenum content adjusted for fission product molybdenum generation, WDS generates compositions of U-8.3Mo and U-9.2Mo for the metal phase, and U-9.7Mo and U-9.8Mo for the interaction product.

(U-Mo)Al_x. The correlation coefficient (R^2) for this equation is 0.974. The swelling terms are defined in the usual manner, $S = \Delta V/V_0$. The volume increase due to fission swelling of the U-Mo-Al solid solution phase ΔV_f , is determined by difference, using equation 7:

$$(7) \quad \Delta V_{\text{U-Mo-Al, fission}} = \Delta V_m - \Delta V_{(\text{U-Mo})\text{Al}_x, \text{formation}} - \Delta V_{(\text{U-Mo})\text{Al}_x, \text{fission}} - \Delta V_{\text{U-Mo-Al}_q, \text{formation}}$$

S_2 , the swelling of the aluminide phase, was based on UAl₃ fuel particle swelling data, [6] fit to a linear equation 8:

$$(8) \quad S_2 = 0.033 * FD,$$

where FD is the fission density in the aluminide phase in units of $10^{21}/\text{cm}^3$.

Consideration of conservation of aluminum and U-Mo alloy mass leads to eqns. 9 and 10.

$$(9) \quad Z = A - Y$$

$$(10) \quad q = (B - W - xY)/Z$$

Equations 3-10 can be solved simultaneously using metallographic and dimensional measurements from Table 2 to yield values for q , x , and S_1 . These values are converted to wt.% aluminum in the (U-Mo)Al_q phase and at.% aluminum in the (U-Mo)Al_x phase, and listed in Table 5. These values provide an estimate of the composition of the fuel and interaction product.

Table 5. Calculation of q , x , and S_1 for RERTR-3 fuel plates.

Plate ID	S_1	q	x	% aluminum	
				(U-Mo)Al _q (wt.%)	(U-Mo)Al _x (at.%)
V03	0.14	0.26	4.8	3.2	83
V07	0	0.27	4.4	3.4	81
S03	0.08	0.40	4.3	4.8	81
E01	0.04	0.23	5.4	2.5	84

It can be seen that the values in Table 5 agree roughly with WDS data in Table 4; however, there are three potential limitations to this analysis. First, the equations relating U-Mo-Al solid solution and (U-Mo)Al_x density to aluminum content are not based on empirical data, since these data are not available for U-Mo-Al ternary phases. Second, the meat volume change is approximated by an average of plate thickness measurements over the fuel zone. This is done because volume measurement by immersion does not yield accurate results on these small specimens. Thirdly, volume fraction measurements were taken from a single cross section of the fuel plate; the measured volume fractions are from a local area assumed to represent the microstructure of the entire specimen. Since the fuel distribution in the plates may not be strictly uniform, some error is introduced by choosing a random section that is assumed to be representative of the entire fuel plate. An estimate of the error introduced by this assumption was made based on

point counting measurements of irradiated plates and unirradiated sibling fuel plates. Measurement of U-10Mo specimens, for example, indicate with 95% confidence that the mean fuel loading is $51.9 \pm 3.3\%$; the same confidence interval applies to the irradiated fuel. In order to gauge the sensitivity of the analysis to uncertainties within this confidence interval, initial volume loading and final measured volume of the phases were varied over a range of $\pm 6\%$. The intercept of the $(U-Mo)Al_x$ density relation (eqn. 6), measured fuel meat swelling, and the amount of initial porosity were also varied. Results are shown in Table 6 for plate V03. Although the resultant values of x and q fluctuate with changes in the input parameters, the incorporation of aluminum in solid solution with the fuel remains necessary to maintain a mass balance within the assumptions given above. The range of q values from 0.2 to 0.4 represents 2.5-4.8 wt.% aluminum in solution.

Table 6. Sensitivity of analysis to input parameters for U-10.6Mo plate V03.

Input parameters ¹						Analysis result ²			
V_o	initial porosity	meat swelling	$V_{(U-Mo)Alq}$	$V_{(U-Mo)Alx}$	eqn. (6) intercept	x	q	Y	Z
51.9	2	0.143	59.5	23.5	9.38	5.0	0.26	0.33	3.9
54.9	2	0.143	59.5	23.5	9.38	5.6	0.20	0.27	4.3
48.9	2	0.143	59.5	23.5	9.38	4.1	0.35	0.42	3.6
51.9	0	0.143	59.5	23.5	9.38	3.8	0.30	0.45	3.8
51.9	4	0.143	59.5	23.5	9.38	5.7	0.24	0.27	4.0
51.9	2	0.08	59.5	23.5	9.38	5.5	0.33	0.27	4.0
51.9	2	0.2	59.5	23.5	9.38	6.0	0.27	0.25	4.0
51.9	2	0.143	56.1	24.9	9.38	5.6	0.21	0.29	4.0
51.9	2	0.143	63.1	22.1	9.38	3.9	0.34	0.42	3.9
54.9	2	0.143	56.1	23.5	9.38	6.2	0.14	0.22	4.3
54.9	2	0.143	56.1	22.1	9.38	6.4	0.14	0.19	4.3
51.9	2	0.143	59.5	23.5	10.3	6.4	0.24	0.27	4.0
51.9	2	0.143	59.5	23.5	8.4	5.8	0.39	0.19	4.1

¹Balance is aluminum. ²Solutions in the range $x=1.2$ to $x=7$.

ALUMINUM DEPLETION RATE in U-Mo ALLOY FUELS

Matrix aluminum depletion for U-10Mo dispersions, normalized to fuel meat fission density, is plotted as a function of temperature in Figure 2. Also plotted are the aluminum depletion predictions for U-Mo fuel based on the correlation derived for U_3Si_2 -Al dispersion fuel and modified for low temperature irradiation of U-Mo/Al dispersions. [7,8] Although the correlation adequately describes the data from the RERTR-1 and -2 experiments, it under predicts the aluminum depletion in U-Mo dispersions at higher temperatures. There are three possible reasons. First, WDS data and calculations based on image analysis point to matrix aluminum depletion by dissolution of aluminum in the alloy fuel particles at high temperature. The U_3Si_2 -Al correlation does not account for this. Second, WDS data suggest that the interaction layer is aluminum rich, with a bulk composition of $(U-Mo)Al_{4+}$. Formation of an aluminum rich layer would result in additional aluminum depletion from the matrix. Lastly, the peak fuel meat temperature may be higher than the beginning-of-life temperature reported in Table 1. If a significant amount of interaction layer were to form early during irradiation, the fuel meat

temperature would rise due to decreased fuel meat thermal conductivity. This would, in turn, result in faster fuel/matrix interdiffusion and a further decrease in thermal conductivity. Higher temperature would shift the points shown in Figure 2 to the left. An attempt is being made to model the effect of fuel-matrix reaction on the thermal history of the fuel plates.

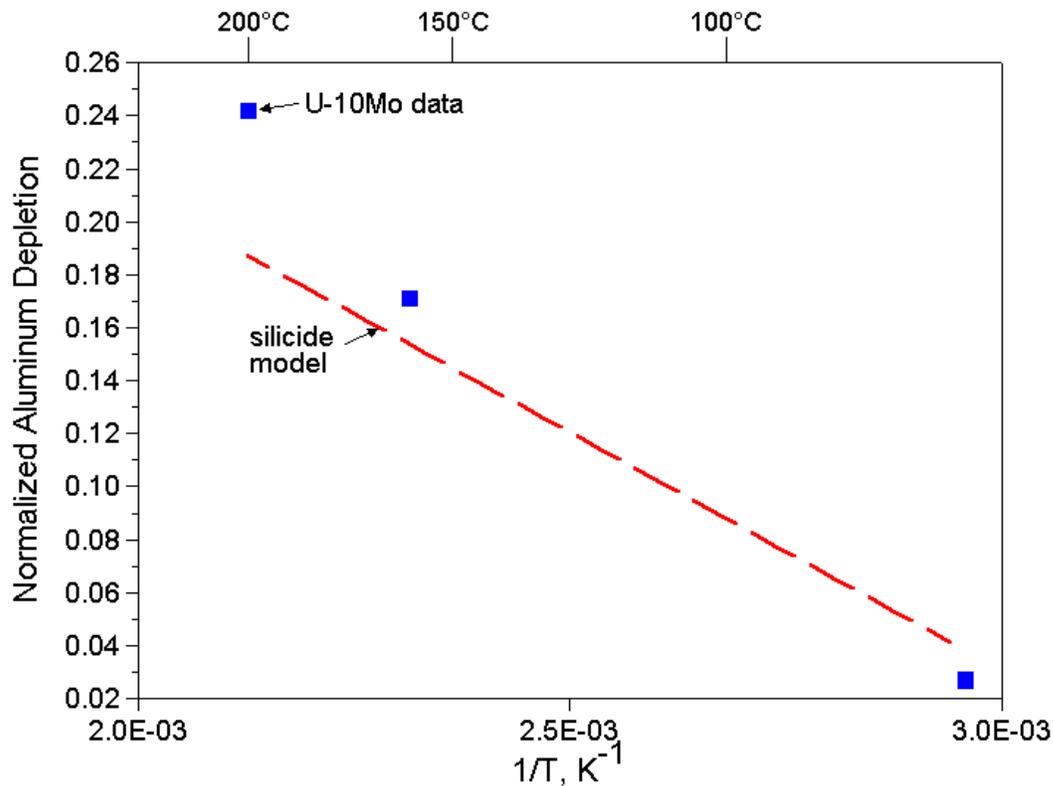


Figure 2. Comparison of aluminum depletion as a function of temperature. Normalized for fuel loading and fuel meat fission density ($\text{cm}^3/10^{21}$).

IRRADIATION BEHAVIOR OF INTERACTION LAYER

Although it is not possible to provide a quantitative description of fuel-matrix interaction kinetics from the results to date, there are certain general conclusions that can be drawn. The first is that composition plays an important role in interaction behavior. In general, the higher alloy fuels react more slowly than lower alloy fuels. This can be seen by comparing data in Table 2 for U-10.6Mo fuel plate V03 and U-6.7Mo plate S03. Plate S03, with a BOL temperature of 204°C consumed more aluminum and formed more interaction product than did plate V03. It is also evident that the small addition of osmium to plate E01 did not slow the rate of interaction, and may have increased it.

The consequences of formation of this large amount of fuel/matrix reaction on fuel performance appears to be benign. Within the range of burnup and temperature so far examined, the interaction layer formed on U-Mo plates during irradiation exhibits stable

irradiation behavior. Figure 3(a) shows that at 69% burnup and approximately 60°C, a thin, coherent reaction layer forms. There are a few small bubbles visible within the interaction layer. Figure 3(b) is a lower magnification image of the entire fuel particle, showing a zone inside the interaction layer which shows a lower fission gas bubble density than the interior of the fuel particle. The interaction layer on plate V03, irradiated to 37.6% burnup at a BOL temperature of 196°C, though considerably wider than that of the low-temperature plate, has a microstructure (Fig. 3(c)) similar to that of the low-temperature specimen. Only isolated pockets of gas bubbles are found. Despite nearly complete aluminum depletion in the case of U-Mo-Os plate E01, there is no evidence of separation of the remaining matrix aluminum from the fuel particles, or of the fuel meat region from the cladding (Fig. 3(d)). Complete reaction of an 8 gU/cm³ U-10Mo fuel meat (87 wt.% U-10Mo) to consume all matrix aluminum would lead to an average stoichiometry of (U-Mo)Al_{1.15}, a mixture of UAl₂ and U-Mo-Al solid solution phases.

The irradiation behavior of the interaction layer observed in these tests is comparable to that of UAl_x particulate dispersions. These fuels are mixtures of UAl₂, UAl₃, and UAl₄ and exhibit stable irradiation behavior to high burnup. Fuels containing initially stoichiometric UAl₃ and UAl₂ fuel show similar behavior. [9] During irradiation of high-density UAl_x plates, reaction of aluminide particles with matrix aluminum did not greatly affect swelling behavior. Complete reaction of aluminide phases during irradiation to form a solid UAl_x fuel meat resulted in a volume decrease at 90.5% U-235 burnup. [10] The microstructures of plates fabricated from two different aluminide compositions were irradiated and examined in this work. No fission gas bubbles were visible within the fuel phase of plates fabricated with fuel particles containing 82.7 wt.% uranium (UAl_{1.9}). One plate that initially contained fuel particles with 80.3 wt.% uranium (UAl_{3.25}) showed a three-phase microstructure, with one phase showing preferential fission gas bubble nucleation and growth. Other work has shown that bubble growth in UAl_x compounds is associated with an impurity phase. [11]

CONCLUSIONS

The matrix aluminum depletion rate of fuels irradiated in the RERTR-3 experiment was more rapid than predicted by extrapolation of data from the low-temperature RERTR-1 and -2 experiments. Similar to previous results, depletion rate has a strong compositional dependence. Lower molybdenum alloys react with the matrix at a faster rate than high-molybdenum alloys. No beneficial effect was noted due to the addition of a small amount of osmium to a U-7Mo alloy.

Electron beam microanalysis (WDS) indicates that molybdenum is present in the interaction layer, possibly as a (U-Mo)Al_x solid solution. WDS data indicate that the interaction layer is aluminum rich. WDS and image analysis methods also indicate that a solid solution of U-Mo-Al forms on high-temperature, high-fission rate exposure. Conclusions must be considered qualitative in nature at this time, due to the lack of quantitative WDS data.

Although the consequences of matrix aluminum depletion appear to be benign under the currently established conditions, the RERTR-4 and -5 miniplate experiments currently under irradiation will provide more data about the behavior of U-Mo fuels at higher burnup than the RERTR-3 specimens, and at higher temperature than the RERTR-2 experiment.

Owing to the large amount of fuel-matrix interaction and the small plate size, the complete thermal history of the RERTR-3 plates has not yet been determined. A model is being developed to reconstruct the thermal history of the fuel plates as a function of power history and change in thermal conductivity due to fission and interaction layer growth.

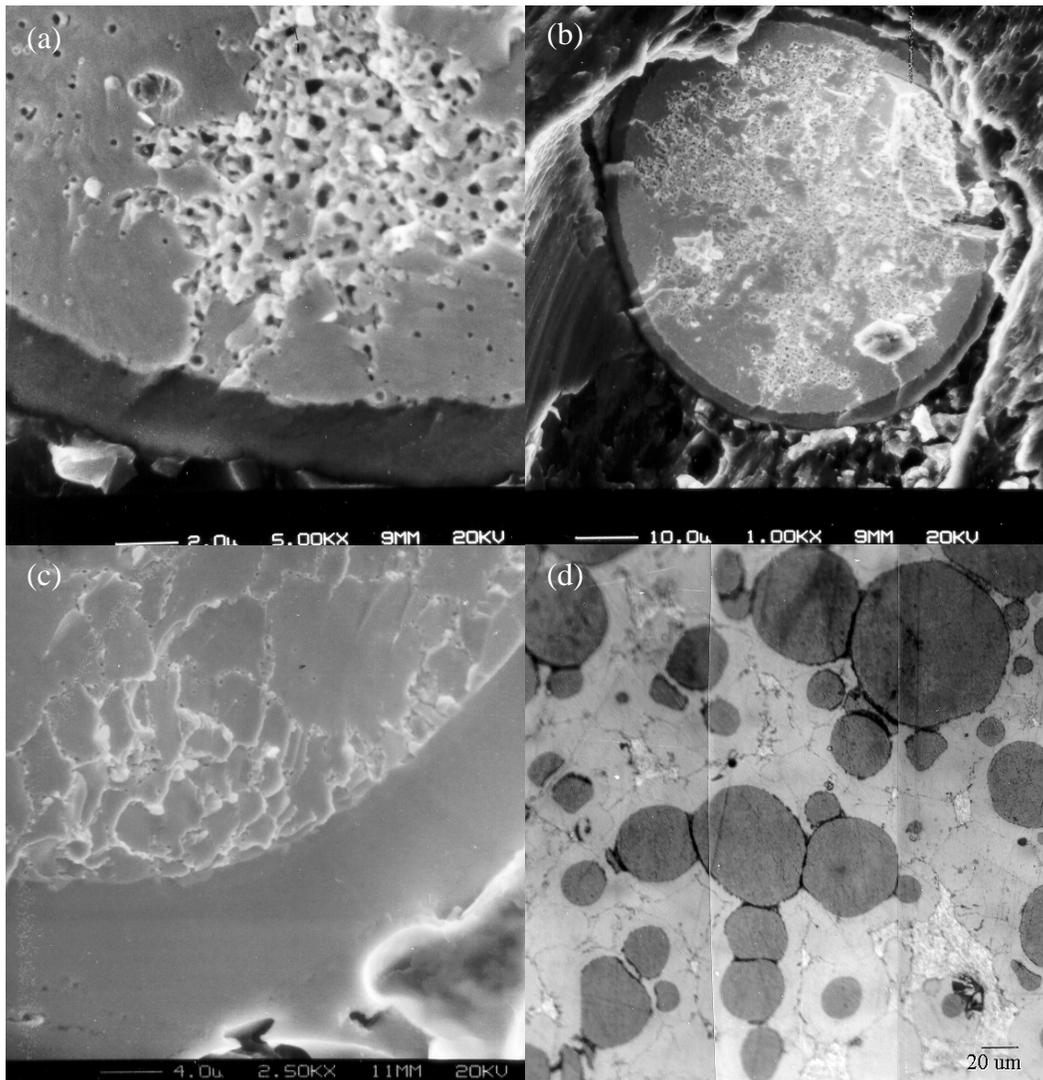


Figure 3. Bubble morphology in $(U-Mo)Al_x$ interaction product in U-10Mo fuel. (a) Specimen V003: 60°C, 5.0×10^{21} fissions/cm³. (b) Same as (a), showing low bubble population near particle periphery. (c) Specimen V03: 196°C, 2.6×10^{21} fissions/cm³. (d) Plate E01. 203°C, 2.8×10^{21} fissions/cm³.

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