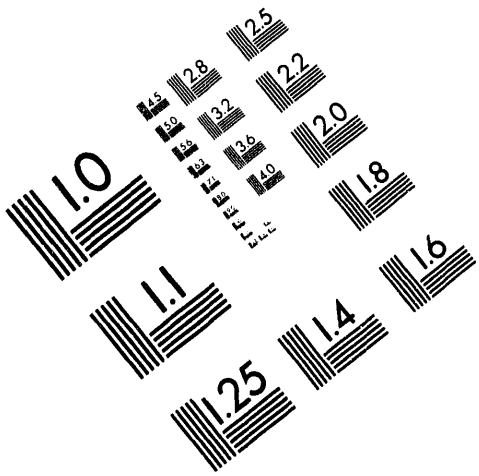

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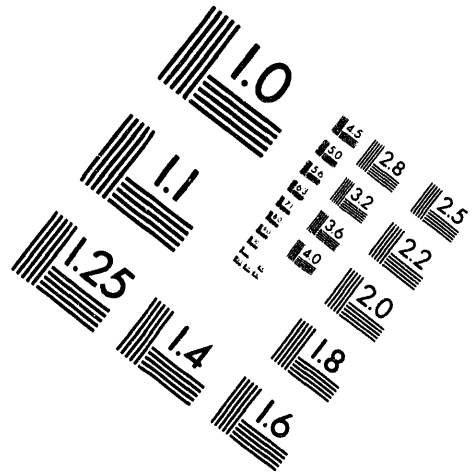

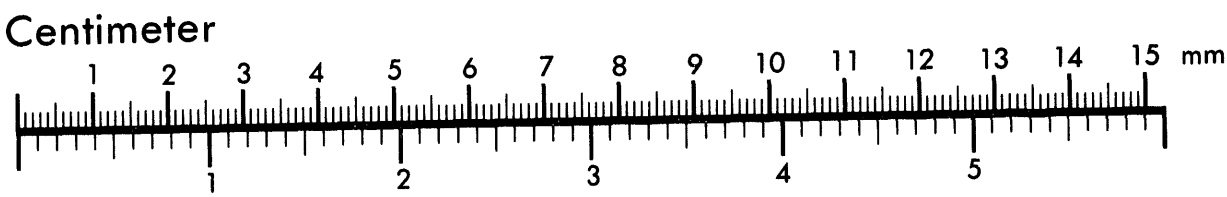

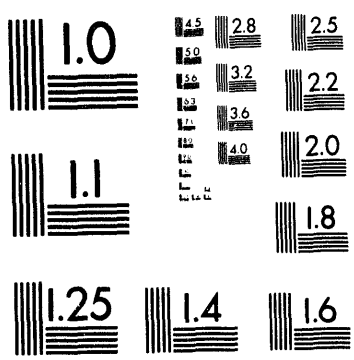

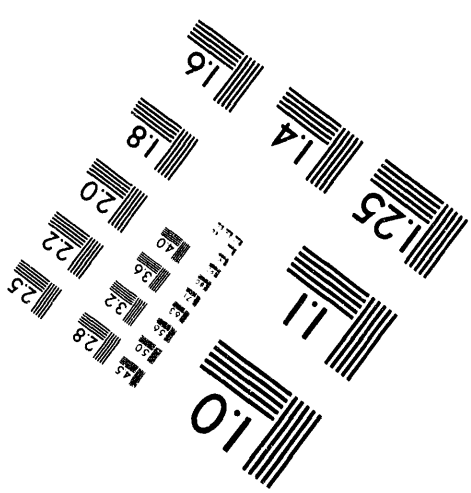

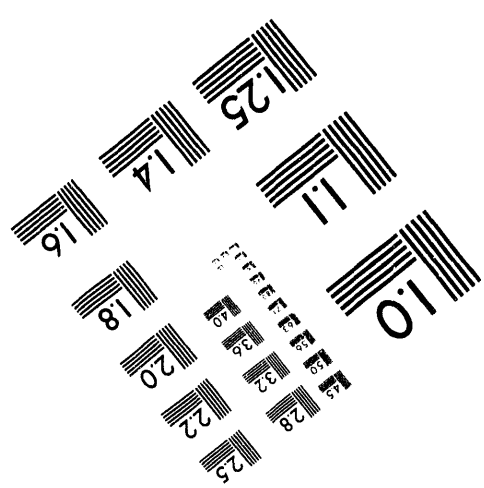

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HANFORD CONTRIBUTION FOR THE 16th HIGH TEMPERATURE FUELS COMMITTEE MEETING MAY, 1963

G. A. Last

I. FUEL ELEMENT SWELLING MODEL - J. W. Goffard

The development of an empirical swelling model by K. R. Merckx and the fitting of the model to KSE-3 fuel element swelling data was discussed in the previous Hanford High Temperature Fuels Committee Report (HW-75288). Fuel swelling data for 14 tube-in-tube fuel elements irradiated to a maximum exposure of 1700 MWD/T has been obtained and compared with computed values using the previously described model. A maximum fuel swelling of approximately 1 per cent was observed in the outer tubular component which was about 25 per cent greater than the maximum observed in the inner component. The calculated and observed swelling data for the 28 components are given in Table I.

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TABLE I

Element	Component	Maximum T, °C	Calculated % ΔV by Model	Measured % ΔV
1	Inner	267	0.12	0.03 (all ±0.15)
2	"	322	0.27	0.40
3	"	371	0.42	0.54
4	"	406	0.76	0.60
5	"	419	1.13	0.54
6	"	424	1.29	0.55
7	"	422	1.19	0.68
8	"	427	1.21	0.77
9	"	430	1.22	0.86
10	"	423	1.02	0.63
11	"	405	0.45	0.55
12	"	374	0.27	0.43
13	"	364	0.21	0.29
14	"	345	0.15	0.09
1	Outer	259	0.15	0.38 (all ±0.10)
2	"	304	0.36	0.54
3	"	344	0.51	0.62
4	"	373	0.63	0.69
5	"	384	0.66	0.72
6	"	389	0.66	0.90
7	"	388	0.60	0.92
8	"	392	0.60	0.77
9	"	396	0.60	1.08
10	"	391	0.54	0.93
11	"	376	0.45	0.58
12	"	352	0.33	0.50
13	"	344	0.27	0.45
14	"	326	0.18	0.47

Measured fuel swelling in the outer components exceeds the calculated values in all cases. The calculated values for the outers are the theoretical minimum swelling since the maximum fuel temperature in the outers did not exceed the temperature at which transition in swelling occurs (about 396 C) as expressed by the model.

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Measured fuel swelling in the inner components is approximately equal to the calculated values for those inner components in which the maximum fuel temperature is less than 400 C, but is less than the calculated fuel swelling in those inner components in which the maximum fuel temperature is greater than 400 C.

The calculated swelling values were made without consideration given to the clad restraint parameter in the model. Comparison of the measured values and the calculated values indicates that clad restraint should be included as a factor in the model, since more swelling was observed in the outer componet than predicted by the model and less swelling was observed in the inner component than predicted by the model. Higher temperature irradiation data needed to evaluate the swelling model further will be obtained from KSE-5 fuel elements currently being irradiated with maximum temperatures to about 600 C.

II. TUBULAR FUEL ELEMENT IRRADIATIONS

Irradiation Performance of Zircaloy-2 Clad Tubular Fuel Elements - J. W. Goffard.

The post-irradiation evaluation of 14 tube-in-tube elements which were successfully irradiated to 1700 MWD maximum exposure is in progress.

The maximum calculated clad strain of the outer circumference of the inner tube based upon O.D. increases is 0.56 per cent, and of the outer circumference of the outer tube is 0.16 per cent. The components appear to be free of any clad striations, bumping or indica-

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tions of clad thinning. Outer tubes have been sectioned and examined in the Radiometallurgy facility. The Be-Zr eutectic brazed end closures show no irradiation effects and no deficienies are apparent. A slight buildup of corrosion product has been observed in a tight crevice between spot-welded supports and the fuel cladding. No appreciable hydriding was observed in conjunction with the corrosion product. The nature of the deposit suggests that the corrosion occurred during autoclave processing and testing and not during reactor operation. The hydrogen content of clad is not significantly different from pre-irradiation levels ($\approx$50 ppm). There have been no crud deposits on the fuel elements and there have been no unusual corrosion behaviors, such as fretting, observed on the elements. In general no deficiencies in irradiation performance were observed.

Dual-Enriched Single Tube Fuel Element - D. P. O'Keefe

The irradiation of an experimental single tube dual-enriched metallic uranium fuel element described in the Hanford contribution to the 15th High Temperature Fuels Committee Meeting was terminated when the element failed during its fifth cycle of reactor operation in the M-3 high pressure water loop at the ETR. The failure resulted from a cladding hot spot which developed in an area where warp of the element had restricted coolant flow. The element had received an exposure of about 5.4×10^{19} fissions/cm^3 (900 MWD/T). A pronounced radial flux gradient exists in this facility and is believed to have been the cause of the warp.

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Fluted Single Tube Element - D. P. O'Keefe

Three irradiation tests are under way to evaluate the fluted tube concept for reactor fuel elements. These elements are Zircaloy-2 clad uranium coextrusions with the outer surface shaped to allow for uranium swelling without tensile elongation of the cladding. This design is intended to avert the localized plastic instability of the outer Zircaloy-2 cladding which has been observed to result from fuel swelling under certain irradiation conditions. It should also reduce the tendency for the inner clad of tubular elements to buckle as swelling develops.

Two large diameter fluted test elements have completed two cycles of irradiation in the ETR-M3 loop at a maximum metal temperature of 520 C. After an exposure of about 2.4×10^{19} fissions/cm^3 (400 MWD/T) no significant volume change was indicated by weight measurements. The irradiation of these elements will continue with interim examinations and volume measurements.

In another test of the fluted geometry a small diameter element has completed six cycles of irradiation in the pressurized high temperature P-7 water loop of the ETR. This element operates with a maximum uranium temperature of about 510 C. After an exposure of approximately 7.2×10^{19} fissions/cm^3 (1200 MWD/T) this element has undergone a volume increase of 3 per cent.

In the third test of this series, three small diameter fluted elements continued their irradiation in cold water.

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III. ZIRCALOY-2 CLADDING STRAIN STUDY - J. W. Weber

Twenty-nine NaK capsules each containing three Zircaloy-2 coextrusion clad uranium rods with intentional striations in the cladding are being irradiated to study the effects of cladding thickness variations on localized plastic straining of the cladding. Eight of the capsules were discharged at a fuel exposure of approximately 6.6×10^{19} fissions/cm^3 (1100 MWD/T) and will be examined shortly. Discharge exposure of the remaining capsules will be based on the Radiometallurgy examination results of the fuel samples in the first eight capsules.

IV. COMPARISON OF SWELLING CHARACTERISTICS OF U-2 w/o ZR AND UNALLOYED URANIUM. - J. W. Weber - NaK capsule irradiations of unalloyed uranium and uranium - 2 w/o zirconium rods clad in Zircaloy-2 have provided a comparison of the fission gas swelling of the two fuels under nearly identical conditions of temperature, exposure, burnup rate and thermal cycling. The volume changes based on density measurements of declad samples, the volume average fuel temperatures, and burnups for the twelve samples are presented in Table II. Volume increases of the unalloyed fuel samples are lower than have been experienced in other capsule irradiations of this type but are probably within the range of uncertainty for calculated temperatures and the density measurements.

Compared with the swelling performance of these unalloyed fuel samples or even with the performance of previous irradiation samples, the U-2 w/o Zr fuel has poorer swelling resistance.

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To learn the reason for this large difference in swelling extensive metallography both optical and electron microscopy is being performed. Comparison of the size and number of fission gas pores in the electron micrographs of the two fuels confirms the differences in swelling as found in the density measurements. (Figure 1) Although the amount of fission gas generated is approximately equal in the two fuels, the agglomeration of the gas into observable pores has occurred extensively in the U-2 w/o Zr but not in the unalloyed uranium.

The heat treatment of the U-2 w/o Zr prior to irradiation (730 C for 5 minutes, quenched in $NaNo_3$ at 350 C for 5 minutes following a 20-second delay in air, and then water-quenched), produced a metastable two-phase structure consisting of approximately equal amounts of α and a higher temperature phase γ_1. Annealing this structure at 650 C for 15 minutes is known to cause the decomposition of the γ_1 phase to $\alpha + \delta$, the δ being an intermediate phase of approximately 70 a/o Zr[1]. Comparison of the pre- and post-irradiation optical and electron micrographs shows that the γ_1, fully transformed to the $\alpha + \delta$ in the center of the fuel but the extent of decomposition decreases toward the cooler edges of the fuel. During irradiation then considerable phase transformation has occurred, with the accompanying movement of phase boundaries. Phase instabilities (decomposition and reversion) during irradiation are thought to have a definite influence on the swelling characteristics of a fuel. This decomposition could be

(1) Angerman, C. N. and W. R. McDonell, Metallography of U-2 w/o Zr Alloy Before and After Irradiation, DP-589.

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one reason for the large amount of swelling in the U-2 w/o Zr. Although annealing studies are under way, it is not known at this time how early in the irradiation thermal decomposition would occur at the irradiation temperatures. Early decomposition, before sufficient fission gas is generated, would minimize the influence of the phase decomposition on swelling. If this is true, then the mere presence of the zirconium atoms in solution in the α-uranium matrix may accelerate the formation and agglomeration of the fission gases.

The presence of the δ second phase seems to have a definite influence on the size and distribution of the fission gas pores. In the hotter regions of the U-2 w/o Zr fuel a zone 5-7μ wide surrounding most of the δ phase regions is essentially void of the smaller gas pores as observed elsewhere in the matrix. (Figure 2) In place of the smaller pores are fewer but much larger gas pores. The range of pore sizes in the matrix is from 0.1μ to 7μ. Near the outer edges of the uranium specimen the zone depleted of small pores is not found surrounding the second phase areas. Diffusion of gas and agglomeration of pores seems to have been sufficient in these cooler regions to give pore sizes and distributions similar to that found in the matrix at the center of the fuel.

In summary, the fission gas agglomeration is much greater in the U-2 w/o Zr fuel than in unalloyed uranium. This is true in both the matrix regions of the alloy fuel where the solubility of Zr in uranium is very small and in the δ second phase regions.

Extensive columnar grain growth was observed in the unalloyed uranium and U-2 w/o Zr. This grain growth has been observed in other irradiations in which it was definitely established that the growth occurred at α phase temperatures. Examples are shown in Figure 3. Some grains appear to originate at or near the center and extend for distances up to one-half the radius. Although most grains are not this long, the elongated nature of the grains extends about three-fourths of the radius on both the alloyed and unalloyed fuels.

TABLE II

FUEL OPERATING CONDITIONS AND MEASURED DENSITY CHANGES

Fuel Sample	Volume Average Temperature (°C)	Fuel Burnup (at %)	Density Decrease (%)
1124	505	.20	8.4
154	505	.19	7.7
244	505	.20	7.7
TC27	560	.22	3.3
TC25	560	.19	3.1
TC23	560	.20	2.3
BB33	545	.20	3.0
BB31	545	.20	2.3
BB212	545	.20	3.5
TB36	565	.18	3.3
TD42	565	.17	2.7
TD45	565	.18	2.7

Unalloyed Uranium

Uranium - 2 w/o Zirconium

Figure 1: Comparison of the Fission Gas Porosity in Unalloyed Uranium and U-2 w/o Zr Irradiated at Comparable Temperatures and Exposures. 5000X

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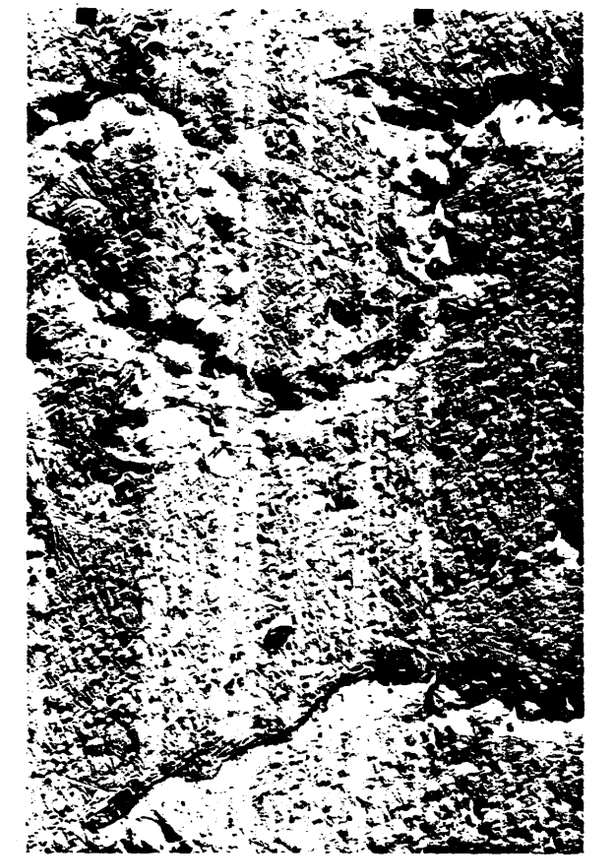
Center of Fuel

Approx. Mid-Radius



Approx. Two-Thirds Radius

Figure 2: Fission Gas Pore Distribution Around Second Phase Particles 5000X

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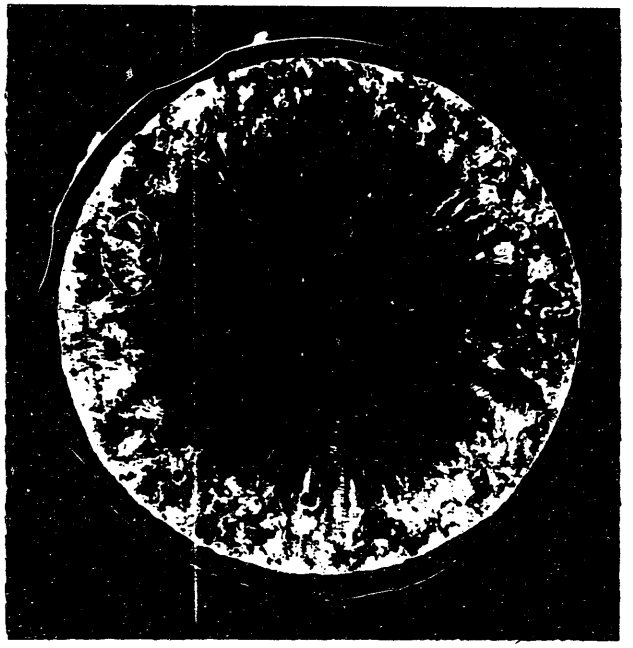
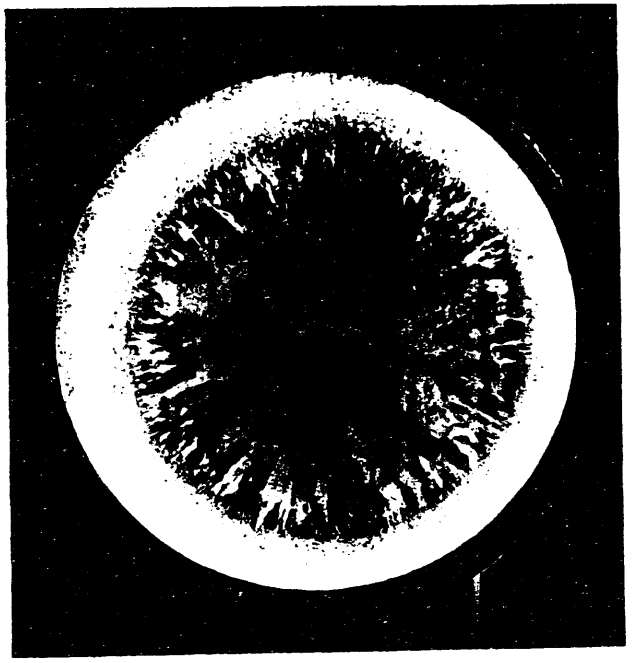


Unalloyed Uranium



U-2 w/o Zr

Figure 3: Columnar Grain Growth in Unalloyed Uranium and U-2 w/o Zr. ~5X

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V. THORIUM-URANIUM ALLOY FUEL - R. S. Kemper

A program has been initiated to evaluate the performance of metallic thorium-uranium fuel under power reactor conditions. Zircaloy-2 clad, fully bonded fuel elements were fabricated by co-extrusion with billets prepared from double vacuum arc melted ingots of thorium-2.5 w/o uranium (93.5% enriched) - 1 w/o zirconium. Bonded end closures were made by brazing Zircaloy-2 end caps in place with Be-Zr-2 eutectic followed by electron beam welding. Fuel elements are 8 inches in length, 1.750 inches O.D., 1.050 inches I.D., with 0.025 inch thick inner and outer Zircaloy-2 cladding. Irradiation testing began in mid-April in a high temperature, high pressure, water-cooled loop in the ETR. The goal of the test program is the attainment of a maximum burnup approaching 3 a/o. During the course of the test the three elements comprising the charge will be subjected to periodic examination and measurement for the surveillance of their behavior. Initial specific power is 195 kw/ft maximum and is expected to decrease to about 150 kw/ft after 1 a/o burnup. The initial heat flux is 1,000,000 BTU/hr-ft^2 on the inner bore surface and will decrease to 750,000 at 1 a/o burnup. The maximum fuel temperature is calculated to be 560 C.

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