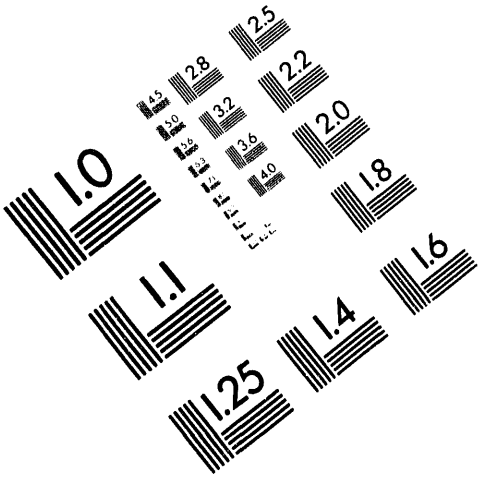

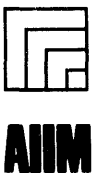

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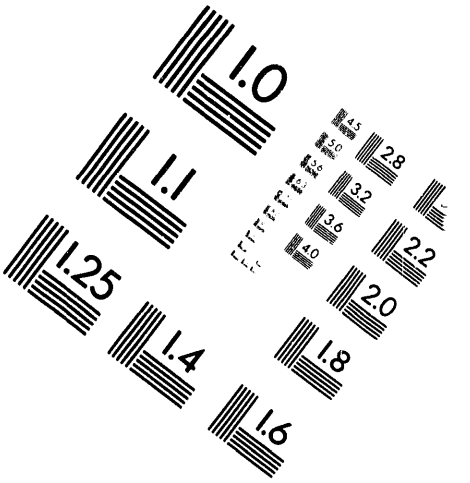

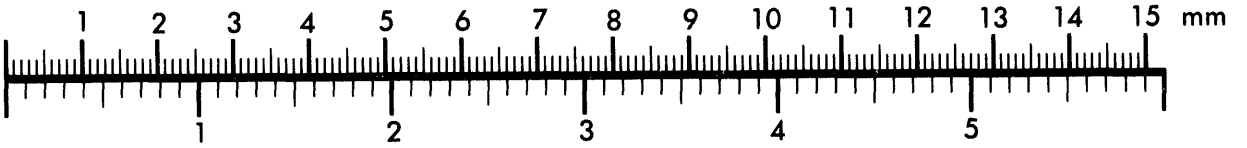

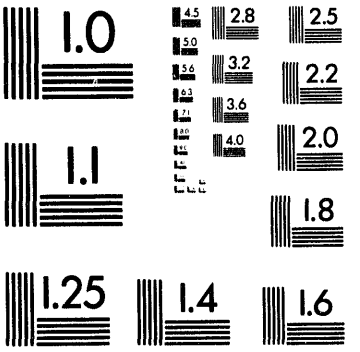

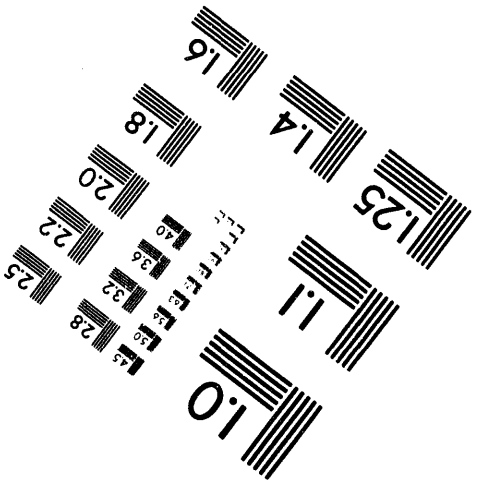

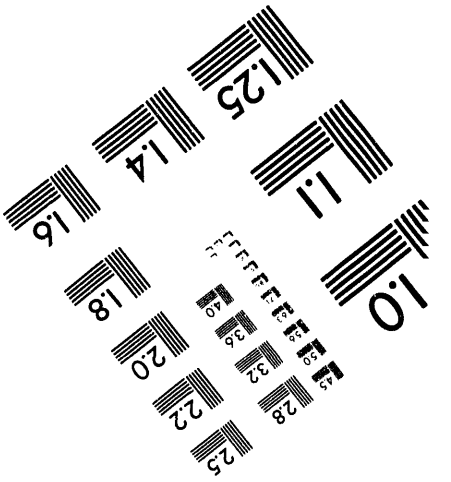

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HANFORD CONTRIBUTIONS FOR THE 16th HIGH TEMPERATURE FUELS COMMITTEE MEETING, November 1962

G. A. Last

March 25, 1962

November 1962

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HANFORD CONTRIBUTION FOR THE 15th HIGH TEMPERATURE FUELS COMMITTEE MEETING, NOVEMBER 1962

G. A. Last

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

Previous analysis of Zircaloy-2-clad tubular uranium fuel element swelling data yielded an empirical expression for the element which relates the total fuel swelling to the 1.65 power of burnup and mean fuel temperature in the temperature range 400-470 C. The expression is:

$$\% \Delta V = 3.0\ (\text{at.}\%\ \text{B.U.}) + 0.0145\ [(Tm-365)(\text{at.}\%\ \text{B.U.})]^{1.65}$$

Tm = volume mean fuel temperature, °C
B.U. = burnup of fuel

This expression describes the fuel behavior during the transition from the theoretical minimum to the theoretical maximum swelling. However, the use of a mean fuel temperature as a parameter in a temperature range where swelling is a highly temperature dependent phenomenon greatly restricts the use of this expression.

A more complete empirical swelling model developed by K. R. Merckx is based upon the temperature, burnup, and pressure dependent swelling behavior of an incremental volume of fuel. The model describes the swelling behavior of an incremental volume of the uranium fuel and can be readily used to describe the swelling of a fuel element by summing of the individual incremental changes in the fuel volumes chosen to represent the element. The model is expressed as follows:

$$v = AB + \frac{GBT}{PC + \Theta_1 B^{\Theta_2} T^{\Theta_3}\ \{1+\exp[-\Theta_4(T+B\Theta_5-\Theta_6)]\}}$$

v = v/o increase of a volume element
A = constant = 3.0

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B = atom percent burnup
G = constant = 550 gas volume generated/cm^3 of fuel/at o/o B.U. x 100
T = $K^o/273$
PC = coolant pressure, atmospheres
Θ_1 to Θ_6 = parameters, statistically evaluated

The values of the parameters have been evaluated to be:

Θ_1 = 19.5067
Θ_2 = -1.1470
Θ_3 = -1.0739
Θ_4 = 195.0000
Θ_5 = 0.11204
Θ_6 = 2.50385

The six parameters of interest were statistically evaluated using a computer program to obtain the best agreement between the model and the experimentally determined fuel element swelling data. Figure 1 is a plot of volume increase versus fuel temperature in °C in the transition range at a coolant pressure of 1600 psi and for burnups to 0.5 percent in 0.1 percent increments. The temperature sensitivity of the transition in swelling from the theoretical minimum to the limiting maximum is very striking. As burnup is accumulated in the fuel, the temperature at which transition occurs is decreased. It is noted that the figure illustrates the swelling behavior of an incremental volume of uranium and not of a specific fuel element.

The theoretical maximum swelling, the plateaus on the right side of Figure 1 by which the transition curves are limited, have not been established experimentally. To obtain data in the temperature region above the transition region, additional tubular fuel element irradiations are planned. These irradiations will furnish fuel swelling data for higher fuel temperatures and will hopefully verify the maximum swelling limits defined by the fuel element swelling model.

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To be useful, this model should predict the swelling of tubular fuel elements of different diameters and fuel elements of other geometries over a range of temperatures, burnups, and restraint conditions. The six parameters are being evaluated from data obtained on Zircaloy-2-clad uranium rods irradiated in NaK capsules, to compare with the values determined from the tubular fuel elements.

II. TUBULAR FUEL ELEMENT IRRADIATIONS

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

N-inner fuel tubes have been examined in detail in the radiometallurgy facility after attaining exposures up to 2100 MWD/T. N-reactor inner tube prototype elements with iron and silicon additions for the coextrusion billet grain refinement, are currently being examined in radiometallurgy after attaining exposures up to 3000 MWD/T in pressurized loops. The performance of these elements has been excellent with no deficiencies or inadequacies noted in the corrosion behavior of the clad, in the brazed closures, or in the braze heat-affected zones. The inner bores of 5 inner tube elements with iron and silicon additions irradiated to 3000 MWD/T had started to distort, presumably due to inner-clad buckling resulting from fuel swelling. This distortion is visibly perceptible, but is not measurable.

A total of 18 N-reactor fuel element assemblies (36 inner and outer tubular components) were weighed in the KE basin for evaluation of fuel swelling following irradiation in KER loops to exposures up to 800 MWD/T. The fuel swelling observed approximated the minimum theoretical swelling computed on the basis of fuel burnup; thus, indicating that swelling resulting from the agglomeration and expansion of the fission gases was not significant in these test elements.

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

An experimental Zircaloy-2-clad single-tube dual-enriched metallic uranium fuel element, has completed four cycles of high-temperature irradiation in the M-3 pressurized water loop in the ETR. The tube is 2.30 inches OD, 0.025 inch outer Zircaloy-2 cladding, 0.185 inch thick ring of 1.6 percent enriched uranium, 0.290 inch thick ring of natural uranium, 0.025 inch inner Zircaloy-2 cladding, and 1.25 inches ID, with all components bonded together. In addition to evaluating the dual enriched fuel element concept, this test is the first irradiation of a thick walled single tube. The element has accumulated a maximum exposure of approximately 900 MWD/T.

The element was examined after the first and second cycles of operation. After the first cycle, the element had warped 100 mils toward the reactor core. It was reinserted with the direction of warp away from the core, but after the second cycle, it was again found to be warped 100 mils toward the core. Since the M-3 test facility is in the reactor shield, an element irradiated in this facility is subjected to a pronounced radial flux gradient. The resulting diametral temperature gradient in the test element is believed to be the cause of the warp. No corrosion effects or other damage have been observed on the element.

Initially in this experiment, the element operated with the maximum uranium temperature about 570 C. In subsequent reactor cycles the decreases in the general power level of the reactor have resulted in progressively lower power generation rates for this experiment. The element currently is operating with a maximum metal temperature of about 450 C. The goal exposure for the experiment is 2000 MWD/T maximum.

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III. ZIRCALOY-2 CLADDING STRAIN STUDY

Localized thinning of "necking" and ultimate fracture of Zircaloy-2 cladding has occurred on a rod from each of two 7-rod clusters irradiated in pressurized water, and on several rods irradiated under a variety of conditions in NaK-filled capsules. The characteristics of the failures are similar and have been described in the 12th High Temperature Fuels Committee Report from Hanford.

To study the effects of cladding thickness variations on the susceptibility to failure, a program of irradiating Zircaloy-2-clad uranium rods in NaK capsules with intentional striations in the cladding is being conducted. Examination is complete on samples from four capsules recently irradiated to an average uranium exposure of 0.2 at.% burnup. Table 1 lists the conditions of the cladding before irradiation, the irradiation conditions, and the results of measurements made on each sample. The cladding strains are determined from the maximum change in diameter. There are no indications of necking or plastic instability of the cladding as seen in some previously irradiated Zircaloy-2-clad rods, even up to the 3.6 percent cladding strain as on sample 244. Figure 2 shows samples 244, 154, and 1124 which experienced the greatest cladding strains of the 12 samples in this test. The grid system of equal sized squares, chemically etched in the cladding surface, was used to determine if there was any non-uniform cladding strain. There was no observable change in size or shape of the grid squares on any of the samples. Thus, the cladding strain was uniform over the circumference on the unstriated samples. In the striated samples, the uniformity of strain is not so clear as the total strain could be divided among the seven striations. Samples from capsules 3U2-3 and 3U2-5

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show an effect of striations on the maximum strain, that is the thinner the cladding at the striations, the larger the total strain. This effect is not consistent on the samples from the other capsules.

Metallographic examination of sample 1124 with approximately 3.5 percent total cladding strain shows that some localized deformation is occurring in the region of the striation, as evidenced by the uranium directly under the seven striations moving outward from its normal circular contour. A comparison of the cladding-uranium interface for sample 1124 and the unstriated sample 244 are shown in Figure 3. However, the structure of the Zircaloy-2 under the striations shows only a few twins and no elongation of grains. It is apparent then that not all the deformation is taking place at the striations. The observed lack of localized deformation at high cladding strain is in agreement with observations on other test elements irradiated at cladding temperatures of 350 C and above as reported in the Hanford contribution to the 14th High Temperature Fuels Committee.

Fabrication of components for a second irradiation test of 125 Zircaloy-2-clad uranium rods with non-uniform cladding thickness has continued. All phases of assembly of the fuel elements is complete and they have been loaded into the capsules. Approximately 75 percent of the capsules have been filled with NaK. Assembly of several capsules to measure the uranium temperature is proceeding using a mechanical seal.

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TABLE 1

Capsule	Fuel Sample	Fuel Composition	Cladding Thick. In.	Maximum Depth of Striation	Calculated Avg Cladding Temperature C	Total Cladding Strain, %	at.% Burnup of Fuel
3U3-2	TD 45	Unalloyed		0	438	1.36	0.18
	TD 42	Uranium	.025	0.004	438	1.0	0.17
	TB 36			0.008	438	1.3	0.18
3U2-3	BB 212	Unalloyed	.025	0	413	1.0	0.20
	BB 31	Uranium		0.004	413	1.45	0.20
	BB 33			0.008	413	1.54	0.20
3U2-5	TC 23	Unalloyed	.025	0	423	1.0	0.20
	TC 25	Uranium		0.006	423	1.2	0.19
	TC 27			0.010	423	1.53	0.21
3A2-2	244	U-2 w/o	.025	0	402	3.6	0.20
	154	Zr		0.004	402	3.26	0.19
	1124			0.008	402	3.46	0.20

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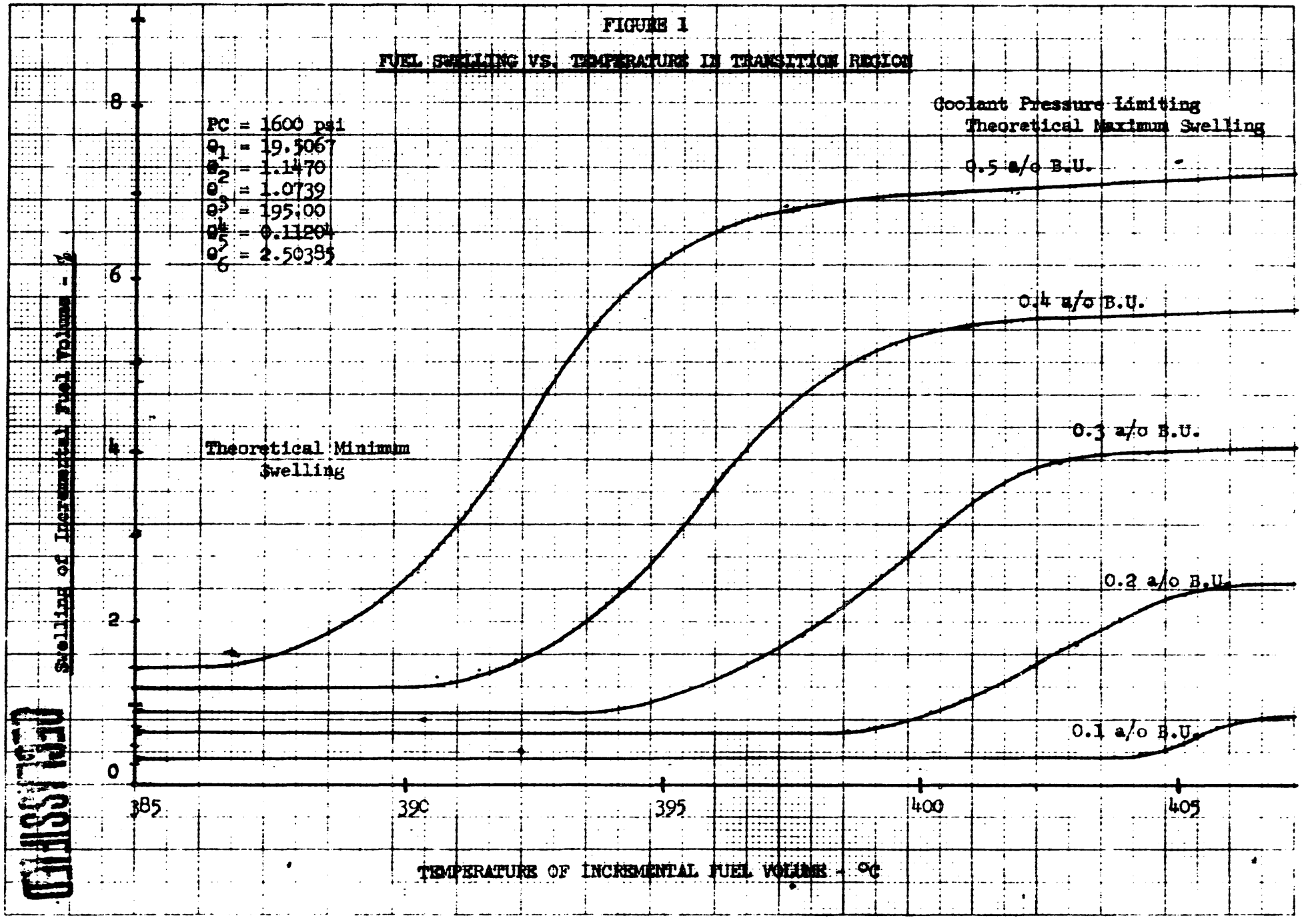
FIGURE 1

FUEL SWELLING VS. TEMPERATURE IN TRANSITION REGION



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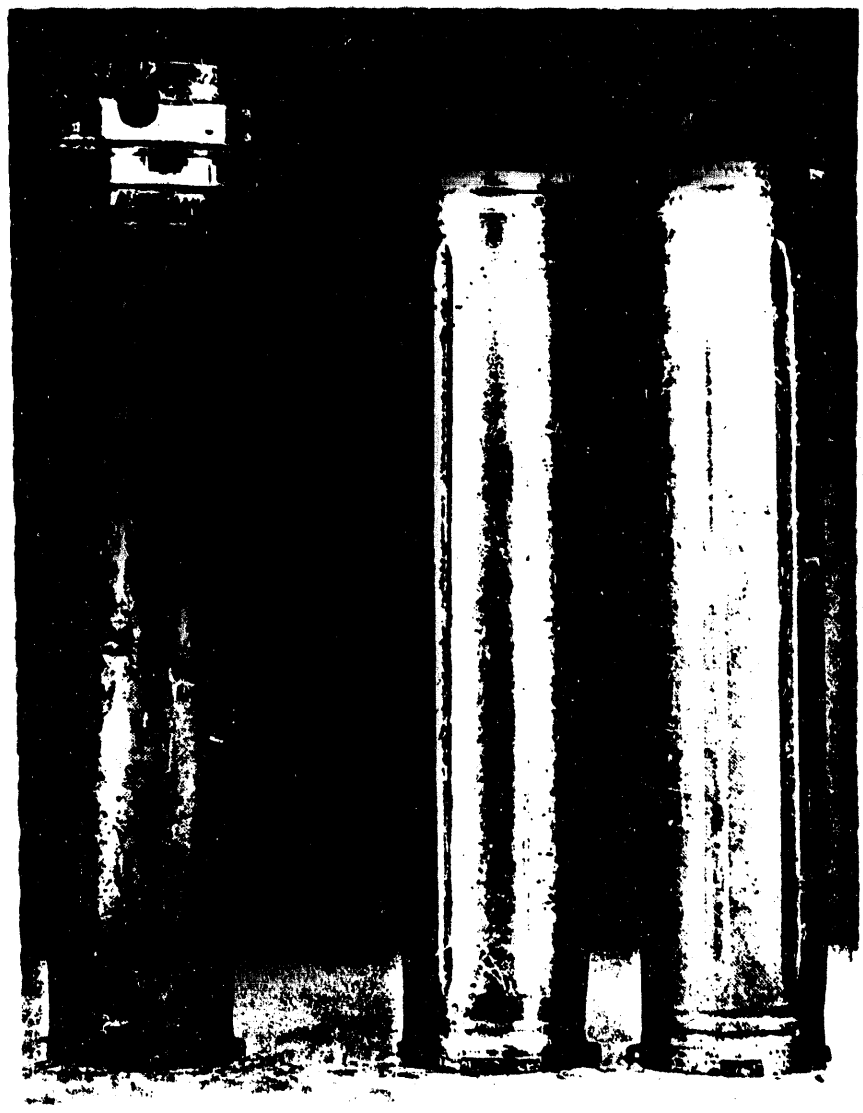

244 154 1124

Figure 2

Samples from Capsule 3A2-2 after irradiation. Striation in Samples 154 and 1124 were machined in the cladding before irradiation to determine the effect of cladding thickness variations on cladding stability.

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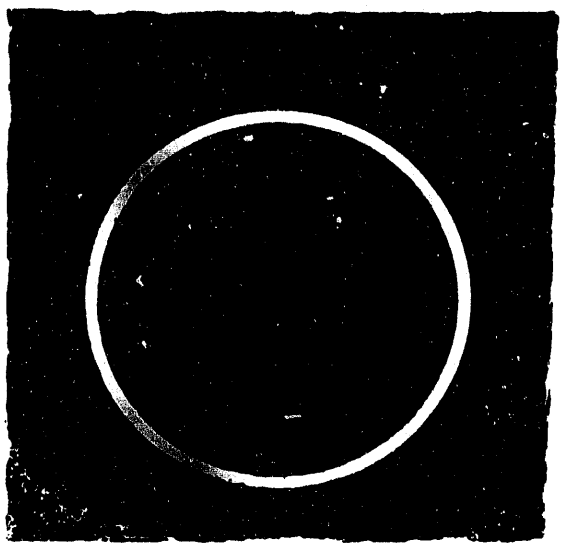
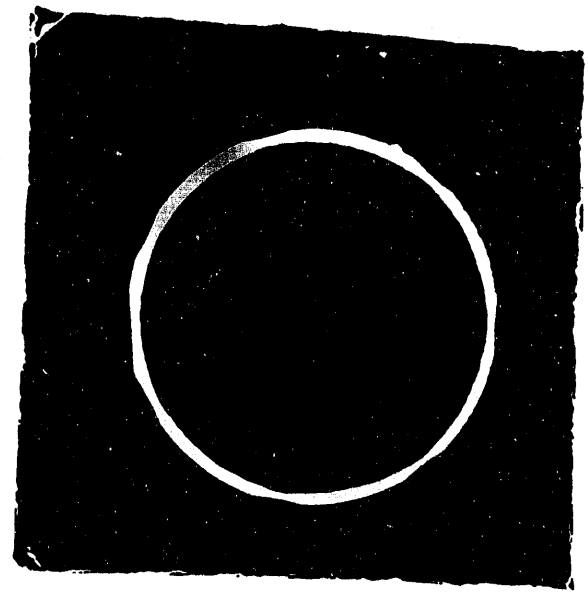


Sample 244
No Striation



Sample 1124
32% Striation

Figure 3

Comparison of postirradiation interface of striated and unstriated samples to show the movement of the uranium outward from its normal contour at the striation.

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