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HANFORD CONTRIBUTION FOR THE NINETEENTH HIGH TEMPERATURE FUELS COMMITTEE MEETING - November 16, 17, 18, 1964 - Los Angeles

G. A. Last

October 21, 1964

SUMMARY

The irradiation performance of Zircaloy-2 clad thorium-uranium fuel elements under water-cooled power reactor conditions continues to be excellent. Fuel swelling continues to be no more than that required to accommodate the fission product atoms. Measured fuel swelling is less than one percent at 3.1×10^{20} fissions/cc (9300 MWD/T).

The addition of iron and aluminum to metallic uranium fuel elements has been shown to markedly decrease fuel swelling. The improved swelling performance is attributed to a decreased tendency to form mechanically induced porosity and is believed to result from the dispersed second phase particles inhibiting the movement of structural defects formed as a consequence of the anisotropic growth of alpha uranium.

HANFORD CONTRIBUTION FOR THE NINETEENTH HIGH TEMPERATURE FUELS COMMITTEE MEETING

Irradiation Testing of Thorium-Uranium Alloy Fuel Elements - R. K. Marshall, J. W. Goffard

Three test elements of Th-2-1/2 w/o U-1 w/o Zr alloy fuel continue to show excellent irradiation performance in the ETR-P-7 high-temperature, pressurized-water loop. The three tubular test elements are coextruded Zircaloy-2 clad, 1.75 inch (4.45 cm) O.D., 1.05 inch (2.67 cm) I.D., 8 inch (20.32 cm) long, and closed with Zr-Be eutectic brazed Zircaloy-2 end caps. The current maximum exposure is 9300 MWD/T (3.1×10^{20} fissions/cm^3). The maximum fuel swelling as determined by interim bulk density measurements is 0.9 v/o. This observed volume change is no more than that required to accommodate the fission product atoms. Visual examination of the elements continues to show excellent performance without indications of clad or closure corrosion.

Table I summarizes the irradiation history to date in terms of the maximum conditions. Fuel swelling behavior vs. fuel exposure and maximum fuel operating temperature vs. fuel exposure are portrayed graphically in Figure 1.

Table I - Irradiation History of Tubular Thorium-2.5 w/o Uranium-1.0 w/o Zirconium Fuel Elements.

Maximum Operating Conditions

ETR Cycle	BU Fissions/cm^3 (MWD/T)	Temp. C	Surface Heat Flux cal/sec-cm^2 (BTU/hr-ft^2)	Specific Power watts/gram (kw/ft)	Fuel Volume Increase, Percent
54	2.8×10^{19} (800)	585	75 (1.0×10^6)	69 (210)	0.15
55	5.4×10^{19} (1550)	538	69 (9.0×10^5)	62 (188)	0.15
56	8.6×10^{19} (2475)	523	65 (8.6×10^5)	59 (180)	Not Weighed
57	1.2×10^{20} (3400)	585	75 (1.0×10^6)	69 (210)	0.5
58	Not Irradiated				
59	1.2×10^{20} (3500)	460	52 (6.9×10^5)	46 (141)	0.5
60	1.6×10^{20} (4500)	540	65 (8.7×10^5)	60 (183)	0.5
61	1.8×10^{20} (5200)	540	65 (8.7×10^5)	60 (183)	0.8
62	2.1×10^{20} (6200)	520	62 (8.3×10^5)	57 (174)	0.8
63	2.5×10^{20} (7400)	520	62 (8.3×10^5)	57 (174)	0.8
64	2.7×10^{20} (8000)	480	53 (7.1×10^5)	49 (150)	0.8
65	2.9×10^{20} (8500)	505	59 (7.9×10^5)	55 (167)	0.9
66	3.1×10^{20} (9300)	460	51 (6.8×10^5)	45 (138)	0.7

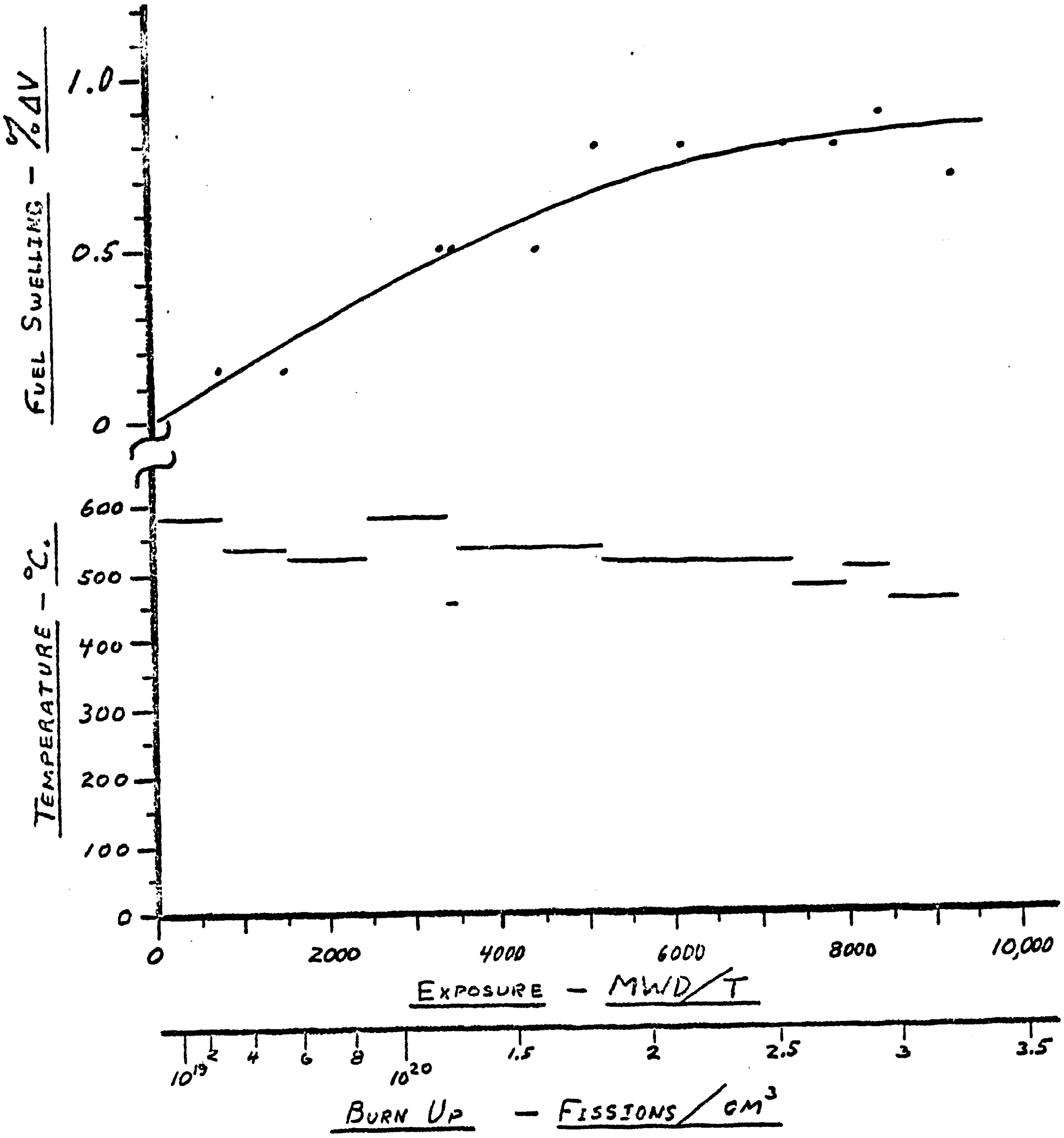

Figure 1 - Swelling Behavior of Th-2-1/2 w/o U-1 w/o Zr Fuel

Comparative Swelling of Uranium Alloy Fuels - J. W. Goffard

The comparative swelling performance of uranium with 120 ppm Si and 140 ppm Fe ("N" fuel composition) and uranium with 400 ppm Fe and 800 ppm Al, has been determined in an irradiation test involving eight experimental fuel elements irradiated in 1350 psi, high-temperature water (KER Loop 2). The test charge, consisting of alternately loaded 12-inch long tubular fuel elements (KSE-5 geometry) of the two compositions, attained an average exposure of 1530 MWD/T (0.92×10^{20} fissions/cm^3). Both fuels were beta-heat treated (730 C for 10 minutes) prior to irradiation. The volume mean fuel temperatures of the test elements ranged from 425 C to 525 C.

Fuel swelling was calculated from pre- and post-irradiation bulk density measurements for each of the elements of the charge. The data shows a significant difference in the swelling behavior of the two fuel compositions with the iron-aluminum additive fuel demonstrating consistently less swelling. Peak swelling of the iron-aluminum containing fuel was 1.2 v/o compared to the peak swelling of the iron-silicon containing fuel of 2.1 v/o. Measured fuel swelling, maximum temperature, and exposure of the individual elements are shown graphically in Figure 2.

Fuel density measurements have been made on declad specimens from two of the fuel elements and the results obtained agree reasonably well with fuel swelling values calculated from the fuel element bulk density measurements. These density data are compared below:

Element	Fuel Swelling from Element Bulk Density	Fuel Swelling from Fuel Density
#5, with Fe and Si	2.12%	1.83%
#6, with Fe and Al	1.15%	0.91%

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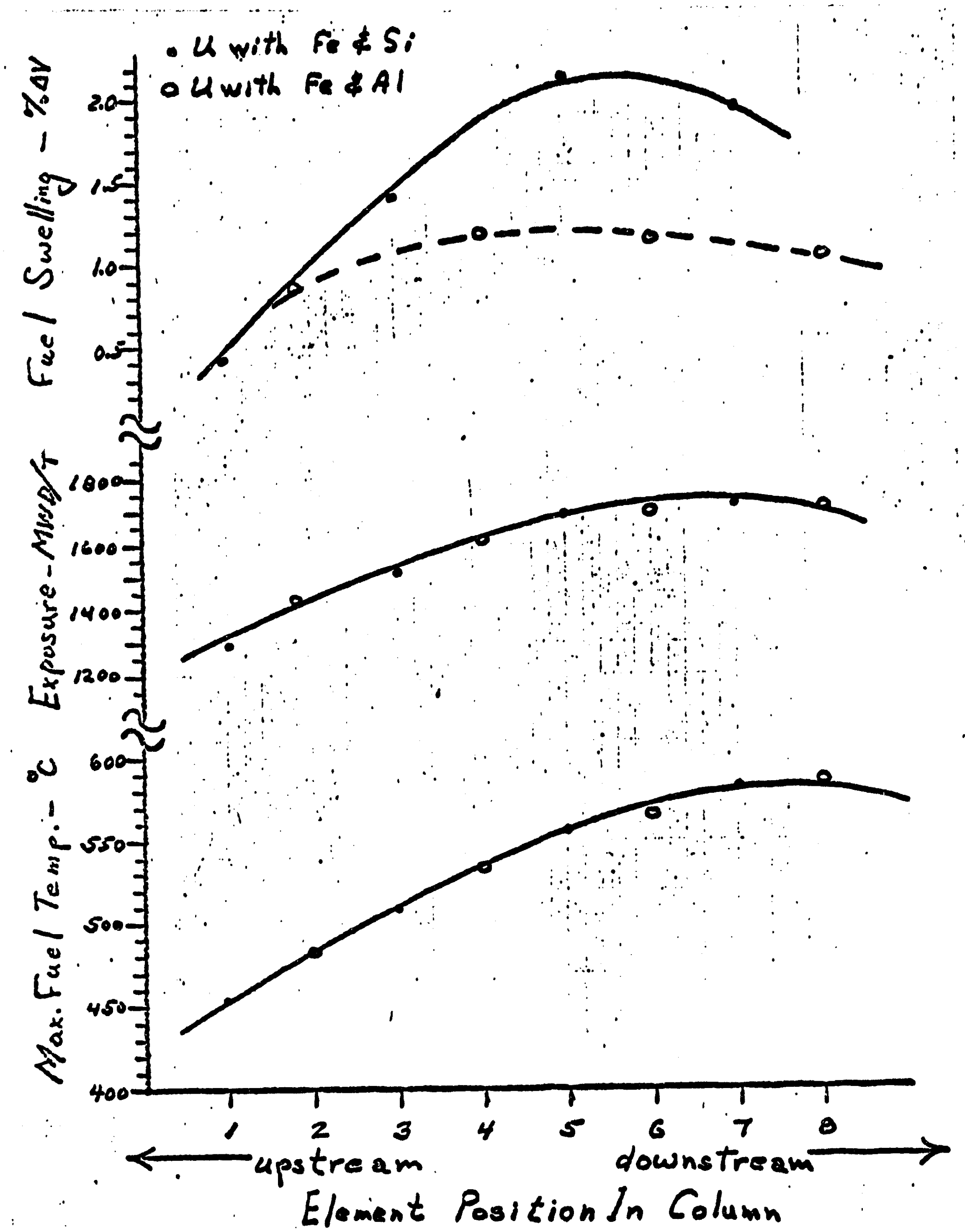

Figure 2 - Fuel swelling, max. temp., and exposure of individual KSE-5 fuel elements in comparative fuel swelling test of uranium alloys.

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One fuel element of each of the two different uranium compositions have been examined in the Radiometallurgy facility. The fuel containing Fe and Si additions contains small voids (grain-boundary tearing) in the fuel adjacent to the cladding and small structurally oriented voids in the central region of the fuel (Figure 3).

The fuel containing the iron and aluminum additions does not display porosity of either type. The cooler regions show a "hashed" or swirled microstructure with no evidence of grain boundary tearing at 15,000 magnification. Likewise, the central region of the fuel showed no porosity at 15,000 magnification. Optical and electron micrographs of the central region Fe and Al containing fuel are shown in Figure 4. Some evidence of re-distribution of second phase precipitate particles present in the fuel was noted in the irradiated Fe-Al additive fuel. The U_6Fe appeared to have coarsened and a network of fine precipitate particles (UAl_2) developed at sub-boundaries.

At the temperature involved in this study, the swelling observed in uranium fuel containing the Fe-Si additions (or fuel with no additives) is largely attributed to "mechanical" damage with the mechanically induced porosity taking the form of grain boundary tearing at the lower temperatures and structurally oriented porosity (emmissary) at the higher temperatures. The mechanism by which the Fe-Al additives prevents the formation of mechanically induced voids is believed to be associated with the ability of the dispersed phase (UAl_2) to inhibit the movement of structural defects, such as dislocations, generated by the anisotropic growth of the alpha uranium grains. Since recovery and recrystallization also involve movement of equivalent structural defects, the recrystallization characteristics of

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<u>Figure 3</u> - Optical micrograph (250X) of fuel containing Fe and Si additions displaying some large porosity and some finer porosity in a twinned structure - 550 C region.

10μ

<u>Figure 3</u> - Electron micrograph (4500X) of same fuel in same region showing spherical porosity.

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Figure 4 - Optical micrograph (250X) of fuel containing Fe and Al additions indicating lack of porosity- 570 C region.

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Figure 4 - Electron micrograph (7500X) of same fuel in same region showing freedom from porosity.

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these fuels was studied.

Specimens of dingot uranium and the two fuels used in this test, each containing 15 percent cold work, were annealed for fixed times at several temperatures and the percent of recrystallization estimated optically. The Fe-Al additive fuel was found to have a 50 percent recrystallization after 20 hours at 575 C, 85 C higher than the Fe-Si additive fuel and 105 C higher than the dingot uranium; thus establishing at least a qualitative relationship between the swelling performance of alpha uranium and the mobility of structural defects.

A second comparative irradiation test of these two fuels has been completed at a 50 percent higher exposure (2500 MWD/T). Although analysis of data are incomplete, the same relative improvement in swelling performance for the Fe-Al additive fuel was observed.

Testing and Evaluation of N-Prototype Fuel Elements - J. W. Goffard

Additional swelling and operational data have been obtained from two charges of prototype N-Reactor tube-in-tube fuel elements which were irradiated in KER loops. The two test charges were irradiated to tube average exposures of 1600 MWD/T ($.96 \times 10^{20}$ fissions/cm^3) and 2000 MWD/T (1.2×10^{20} fissions/cm^3). Maximum fuel swelling observed in the lower exposure test was 0.85 v/o in an inner fuel element and 2.50 v/o in the corresponding outer fuel element. The maximum fuel swelling observed in the higher exposure test was 1.33 v/o in an inner and 3.64 v/o in an outer. Data from these two charges was used in the development of the fuel swelling expression discussed in the previous report(1).

(1) HW-81579 - Hanford Contribution for the Eighteenth High Temperature Fuels Committee Meeting, May 19-21, 1964, (Confidential).

Radiometallurgical examination has been completed on an inner and an outer component from the high exposure test. The exposure and volume increase of the inner component were 2400 MWD/T (1.47×10^{20} fissions/cm^3) and 1.17 v/o respectively. The exposure and volume increase of the outer component were 3070 MWD/T (1.85×10^{20} fissions/cm^3) and 3.64 v/o respectively. Superficially these components were in good condition without indications of warp, clad bumping or rippling, crud deposits, or corrosion effects. Metallographic examinations of the fuel in the outer component (3.64% ΔV) showed that grain boundary tearing was the primary source of volume expansion. Metallographic examination of the fuel in the inner component (1.17% ΔV) at magnification shows grain boundary tearing similar to that observed in the outer component, except that the porosity is finer and can be resolved only with electron microscopy.

Dimensional Changes of Prototype N-Reactor Fuel Elements - J. W. Goffard

Measurement of dimensional changes in N-Reactor fuel element components is part of the program to evaluate the irradiation performance of N-Reactor fuel elements.

Most of the components of two KER loop charges of prototype N-fuel elements have been measured both pre- and post-irradiation. The outer fuel elements are measured at each of ten stations. At each station measurements of the deviations of the wall thickness and the outside radii from a standard are made. The stations are 1.46 inches apart and start and end 5 inches from the ends of the 24-inch long component. The percent change in the cross-section of the outer component at each station was calculated from the pre- and post-irradiation measurements data. The results from the

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higher exposure test are graphed in Figure 5 as a function of the station and the fuel element position in the irradiation test charge. The measurements data shows a significant swelling gradient in many of the individual components. The average swelling of the outer fuel element components, calculated from the wall thickness measurements and length changes are also shown in Figure 5. The calculated fuel swelling is in good agreement with the swelling data as determined from pre- and post-irradiation bulk density measurements. The same type of data and a similar comparison of swelling based upon dimensional measurements and densities was made for the inner components. However, the agreement was not as consistent because of the relatively low precision of the wall thickness measurement obtainable for the inner component on the post-irradiation measurement machine (±0.002 inches) and the smaller volume expansions.

The percent change of the average value at each station in the outside and inside radii was calculated for the outers, and the changes are graphed in Figure 6 as a function of the components' position in the irradiation test charge. The maximum increase in the outside radius of an outer component was 0.007 inch (0.6%) while the maximum decrease in the inside radius of an outer component was 0.004 inch (0.5%). A maximum change of 0.006 inch was noted in the outside radius of the inner component.

Maximum "single-throw" warp calculated from the measurements over an 18-inch span were 0.016 inches and 0.012 inches for the outers and inners respectively. For the outer component, the magnitude of the change in the warp vector appeared to increase with increased fuel swelling.

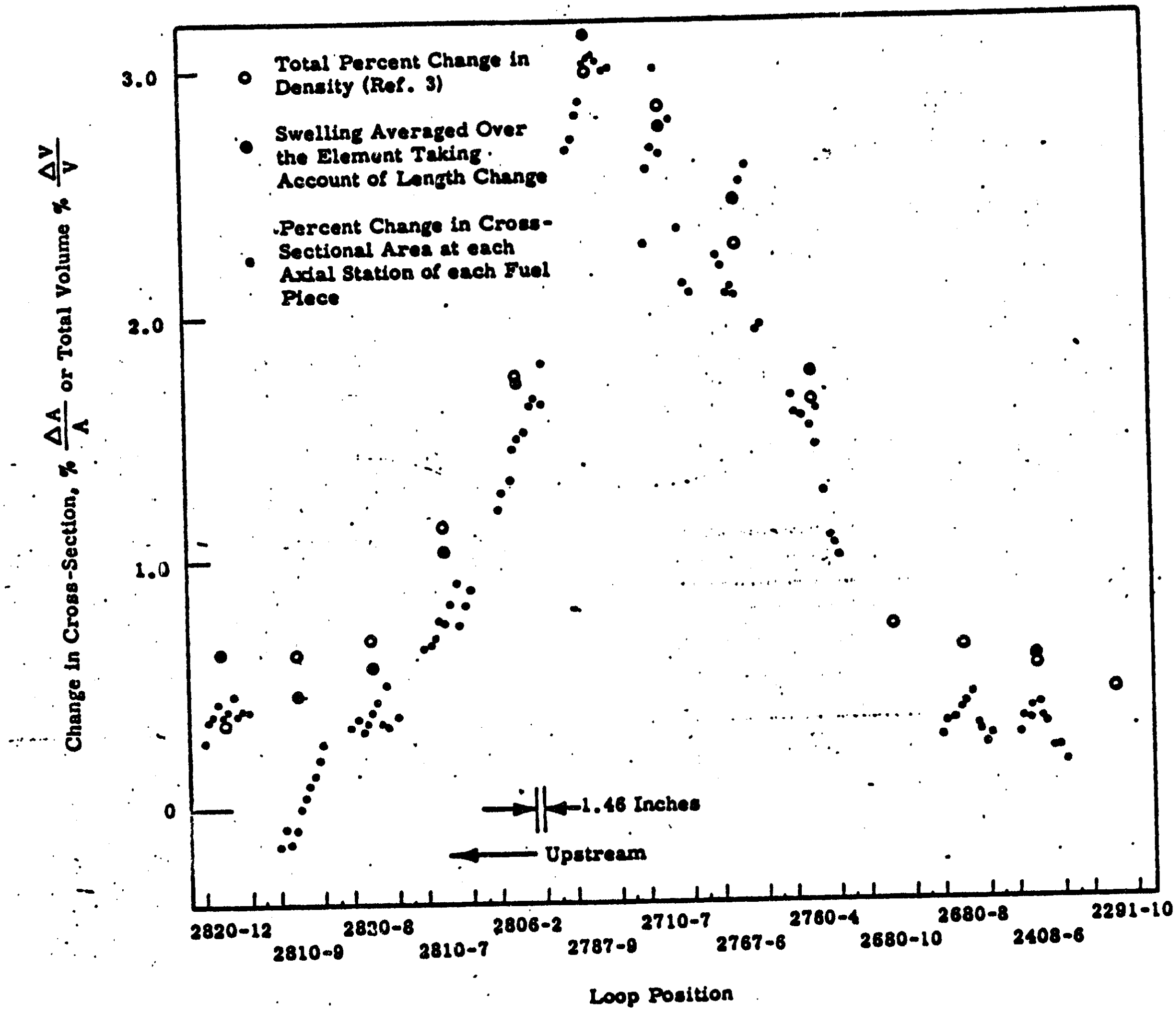

Figure 5 - Percent change in cross-section areas and comparison of swelling determinations vs. loop position for N-Reactor outer components irradiated to a tube average exposure of 1975 MWD/T (KER facility).

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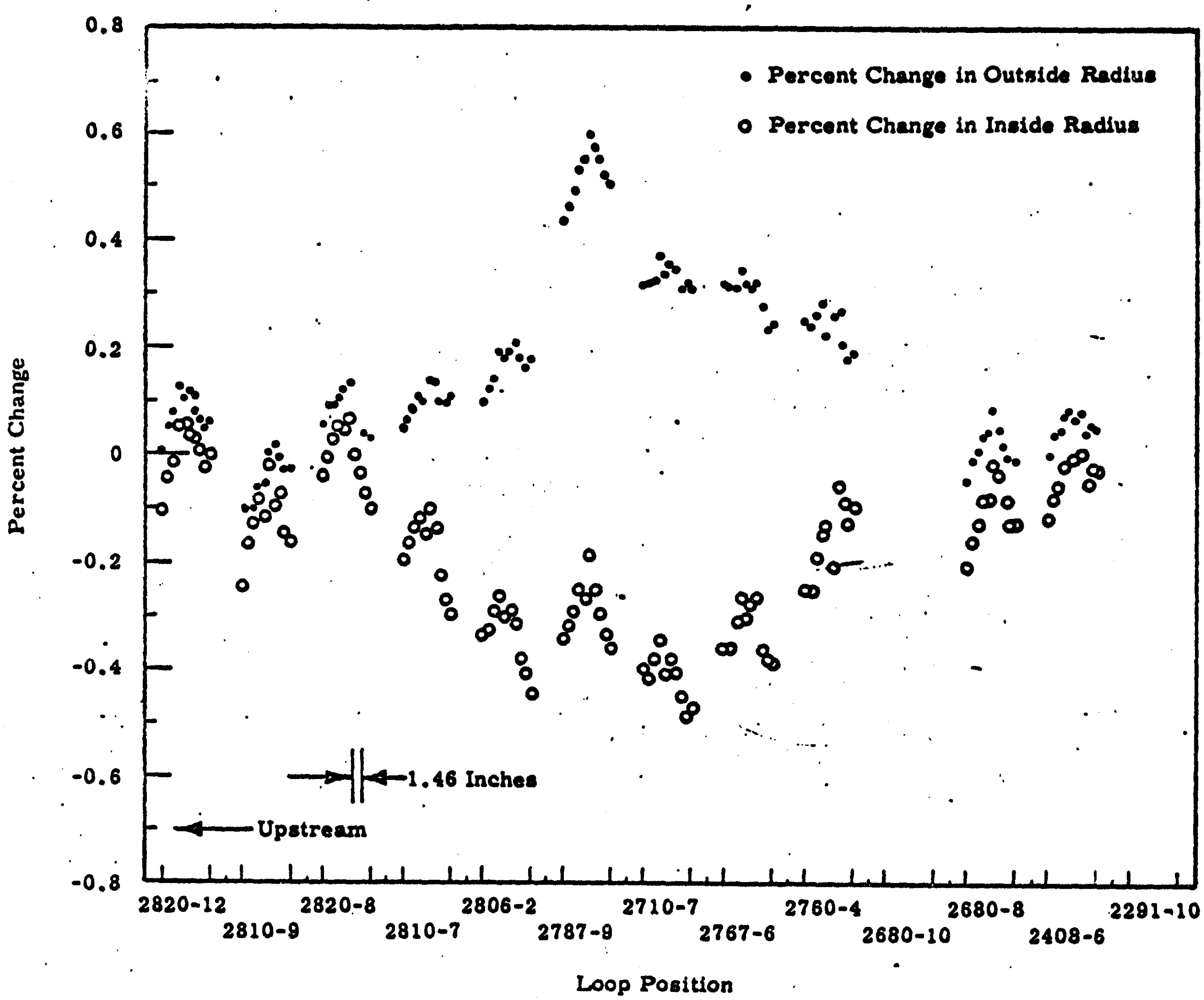

Figure 6 - Percent change in the outside and inside radii at individual measurement stations vs. loop position for N-Reactor outer components irradiated to a tube average exposure of 1975 MWD/T (KER facility).

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Testing and Evaluation of N-Reactor Driver Elements - J. W. Goffard

Thirteen "driver elements" were irradiated in KER Loop 3 to an average fuel exposure of about 1300 MWD/T. The Zircaloy-2 clad driver element is 2.29-inch (5.81 cm) O.D. and 1.44 inch (3.66 cm) I.D. with Be-Zr eutectic brazed end closures. The uranium is the standard "N" composition of 150 ppm Fe and 100 ppm Si.

Fuel swelling of the drivers was determined by pre- and post-irradiation bulk density measurements. A maximum swelling of 3.3 v/o was observed in a driver element in which the fuel is estimated to have operated with a maximum temperature of 530 C. The estimated fuel exposure and the measured fuel swelling for the thirteen driver elements is shown in Figure 7.

Radiometallurgical examination is being conducted on the driver element that experienced 3.3% increase in fuel volume. A considerable amount of grain-boundary tearing porosity is observed in the uranium adjacent to the cladding. The fuel in this region displays a swirled or "hashed" microstructure characteristic of uranium irradiated in the temperature range 350-400 C. The hotter central zone of the fuel (525-550 C) shows heavily twinned grain structure with considerable structurally oriented porosity which is associated with the deformation on twin bands.

Submicron UC Dispersion in Metallic Uranium - R. K. Marshall
J. W. Weber

Metallic uranium fuel rods containing a submicron dispersion of uranium carbide are being irradiated to establish the effect of carbide size on irradiation stability of metallic uranium.

One NaK capsule containing two uranium rods, one rod with fine and one rod with coarse uranium carbide dispersions, has been discharged at

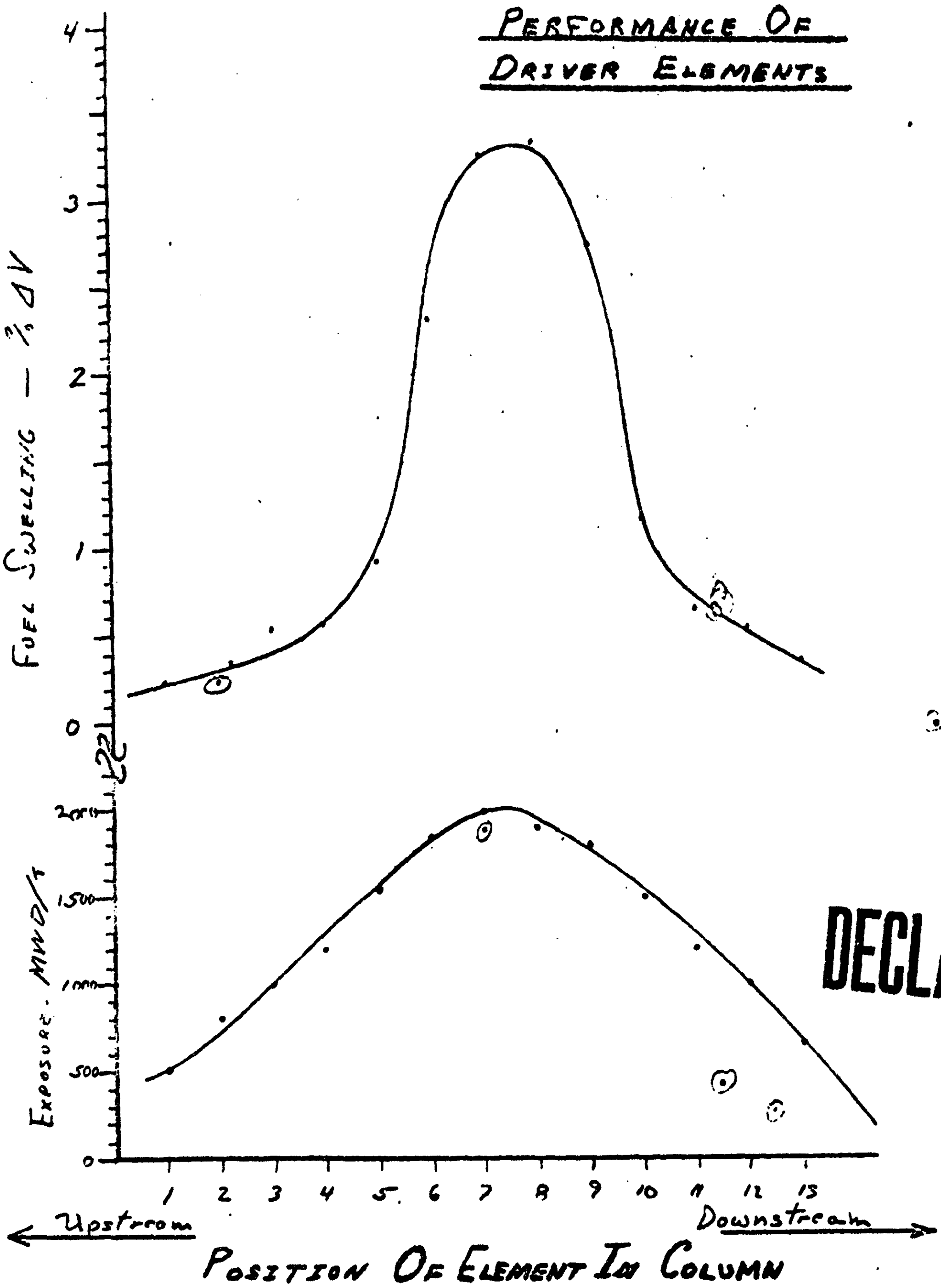

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Figure 7 - Fuel exposure and fuel swelling in Co-Product Driver Element vs. position of element in KER Loop.

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an exposure of 0.3 a/o burnup. The maximum uranium temperature during the last third of the irradiation was 500 C for the fine carbide uranium rod and 450 C for the uranium rod with the coarse carbide. Radiometallurgy examination has been started.

Two other NaK capsules have achieved an exposure of 0.17 a/o at a maximum uranium temperature of 475 to 500 C.

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