DP - 709

METALS, CERAMICS, AND MATERIALS
(TID-4500, 17th Ed.)

IRRADIATION OF A U - 2% Zr FUEL TUBE IN THE VBWR

by

Henry C. Quigley
Reactor Materials Section
Wilmington, Delaware

April 1962

Issued by
E. I. du Pont de Nemours & Co.
Explosives Department - Atomic Energy Division
Technical Division - Savannah River Laboratory

Printed for
The United States Atomic Energy Commission
Contract AT(07-2)-1

DISCLAIMER

ABSTRACT

A coextruded 2-inch-diameter tubular fuel element of uranium - 2% (by weight) zirconium alloy clad with Zircaloy-2 was irradiated in the Vallecitos Boiling Water Reactor to 1280 MWD/T burnup with a maximum core temperature of 433°C. The purpose of this irradiation was to determine the dimensional stability of the alloy under conditions approaching those contemplated for D_2O-cooled power reactors. Successive dimensional measurements revealed a sharp change in swelling rate between 700 and 1280 MWD/T burnup. The amount of swelling increased during this interval from 0.8 to 3.6% (0.7% final cladding strain). This test indicated that the dimensional stability of the U - 2% Zr alloy is inferior to that of unalloyed uranium irradiated under similar conditions.

CONTENTS

LIST OF TABLES AND FIGURES

IRRADIATION OF A U – 2% Zr FUEL TUBE IN THE VBWR

INTRODUCTION

Coextruded tubular fuel elements with Zircaloy cladding and cores of uranium and uranium-base alloys are being developed as part of the Du Pont program on development of heavy-water-moderated, natural-uranium-fueled power reactors[1]. The goal of the over-all program is the advancement of D_2O reactor technology to the point that full-scale plants employing such reactors could produce electricity at costs competitive with fossil-fueled plants. The coextruded tubular element is one of the major fuel element candidates in this program because of its potentially low fabrication cost[2].

A uranium - 2% zirconium alloy core was investigated because of (a) the slightly better corrosion resistance of the alloy compared with unalloyed uranium, (b) the ability to form a corrosion-resistant diffusion barrier between the alloy and the Zircaloy cladding, (c) the excellent compatibility of the alloy and the cladding in the coextrusion process, and (d) preliminary indications of satisfactory irradiation stability.[3]

The purpose of the irradiation in the Vallecitos Boiling Water Reactor (VBWR) was the determination of U - 2% Zr element performance under power reactor conditions. Irradiations of this type of element in Savannah River production reactors involved cladding temperatures and coolant pressures significantly lower than those of interest in the power program. It was particularly desired in the VBWR irradiation to determine the rate of swelling of the U - 2% Zr element at surface temperatures near 300°C.

SUMMARY AND CONCLUSIONS

A tubular uranium - 2% zirconium fuel element clad with Zircaloy was irradiated in the VBWR under power reactor conditions to a burnup of 1280 MWD/T (peak). The element, having a 2-inch OD, $1\frac{1}{2}$-inch ID, 44-inch length, and 0.015-inch nominal cladding thickness, was fabricated by a coextrusion process[4]. During irradiation, the element had a maximum cladding surface temperature of 300°C, a maximum cladding-core interface temperature of 345°C, and a maximum core temperature of 433°C; it underwent approximately 65 power cycles of amplitude greater than 50% of full power. The test was discontinued when the VBWR was shut down for a prolonged period.

Dimensional and volume displacement measurements that were made during shutdowns at 150, 890 and 1280 MWD/T burnup (peak) revealed that the rate of volume change below 890 MWD/T burnup was 6 or 7% increase in volume per atom per cent burnup, whereas the rate of change between 890 and 1280 MWD/T was three times this amount. The 1280 MWD/T measurements indicate a strain in the outside cladding of 0.7%. The inside cladding dimensions changed a relatively small amount, and there was no significant change in length and bow of the element.

If the higher rate of volume change observed between 890 and 1280 MWD/T were to continue, the element life would be limited to about 2200 MWD/T (peak) assuming 2% tolerable strain. If the over-all swelling rate were to prevail, the useful life of the element on the above basis would be limited to about 3800 MWD/T (peak). Savannah River irradiations under somewhat different conditions indicate that the swelling rate decreases at a burnup above that achieved in the VBWR irradiation. However, a concurrent irradiation of an unalloyed uranium element under conditions similar to those of the VBWR test demonstrated significantly better dimensional stability[5]; therefore, with the extended shutdown of the VBWR, investigation of the U - 2% Zr alloy in the Du Pont power reactor program was discontinued.

DISCUSSION

DESCRIPTION OF ELEMENT

Six Zircaloy-clad uranium - 2% (by weight) zirconium tubes were co-extruded by Nuclear Metals, Inc.,[4] for Du Pont as candidates for irradiation in the VBWR. The complete characteristics of these tubes, including results of both nondestructive and destructive evaluation, have been reported by Nuclear Metals, Inc., in a separate report[6]. Pertinent details regarding the element irradiated in the VBWR (Tube No. 41) are reviewed below.

FABRICATION PROCESS

The coextrusion process employed for the test specimens involves the extrusion of a composite billet consisting of an annular U - 2% Zr core, inner and outer Zircaloy sleeves, and front and rear annular Zircaloy end plugs assembled in a copper can with a copper nose plug. The billet, approximately 6 inches in diameter, is extruded over a mandrel through a die at 1200°F on a 2400-ton tube press to produce the final 2-inch-diameter fuel element having bonded cladding and integral end plugs. The VBWR candidates were heat treated for 9 hours at 1630°F in evacuated steel containers to enlarge the corrosion resistant diffusion zone between the cladding and the core, following which the containers were water quenched. The elements were autoclaved in 650°F water for 24 hours and 750°F (1500-psi) steam for 24 hours and were evaluated thoroughly by nondestructive methods.

ELEMENT CHARACTERISTICS

The specification drawing for the VBWR candidate elements is presented in Figure 1. The measured preirradiation characteristics of the in-pile specimen (No. 41) are given in Table I. All characteristics of the element, including autoclave performance, surface quality, bond quality (ultrasonic test), bow, materials analysis, and dimensional variations, met generally accepted standards.

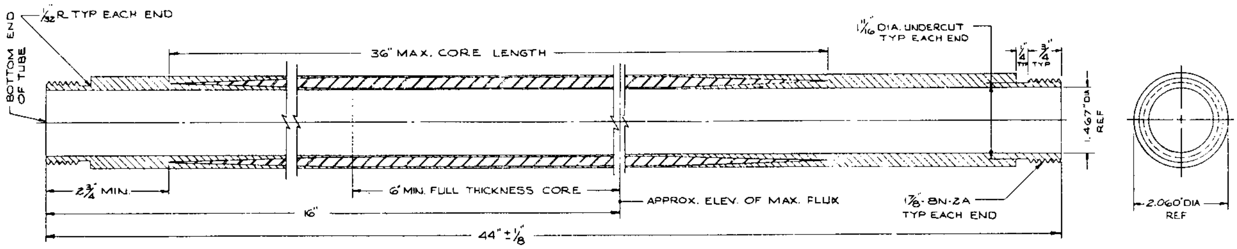

FIG. 1 SPECIFICATION DRAWING OF U – 2% Zr ELEMENT FOR VBWR IRRADIATION

TABLE I

PREIRRADIATION CHARACTERISTICS OF IN-PILE SPECIMEN

Outside diameter of element (average)	2.065 inches
Inside diameter of element (average)	1.474 inches
Outer cladding thickness	
Average	0.016 inch
Minimum	0.011 inch
Inner cladding thickness	
Average	0.016 inch
Minimum	0.013 inch
Length of element (including threaded ends)	44 inches
Core length (tip-to-tip)	34-1/4 inches
Length of core region of uniform thickness	22-3/4 inches
Length of upper tapered region of core	6-3/4 inches
Length of lower tapered region of core	4-3/4 inches
Length of upper solid Zircaloy end	7 inches
Length of lower solid Zircaloy end	2-3/4 inches
Core weight	12.694 kg
Uranium enrichment	3.06% U^{235}
Carbon content of core	198 ppm
Zirconium content of core	1.88 wt %

OPERATING CONDITIONS

The U - 2% Zr tubular element operated in a regular lattice position in the 3 by 3-foot VBWR core and was cooled directly by the VBWR coolant. During the period of the irradiation, March through December 1959, the VBWR was operated under natural-circulation boiling-water conditions(7,8). Because of operating cycle and fuel performance tests being conducted by General Electric during this period, the VBWR underwent considerable variation in power level. The U - 2% Zr element, within its VBWR channel box (Figure 2), was frequently moved from one core lattice position to another in an effort to maintain the element in the highest neutron flux commensurate with a burnout safety factor of 2.0 at 125% of scheduled full reactor power. The element operated at this maximum level, equivalent to 250-kw element power, during approximately 10% of the time at power. The remainder of the significant power operation was largely at 200 kw. Conditions corresponding to these element power levels are given in Table II. The variation of conditions over the length of the element is reflected by the flux and temperature curves in Figure 3.

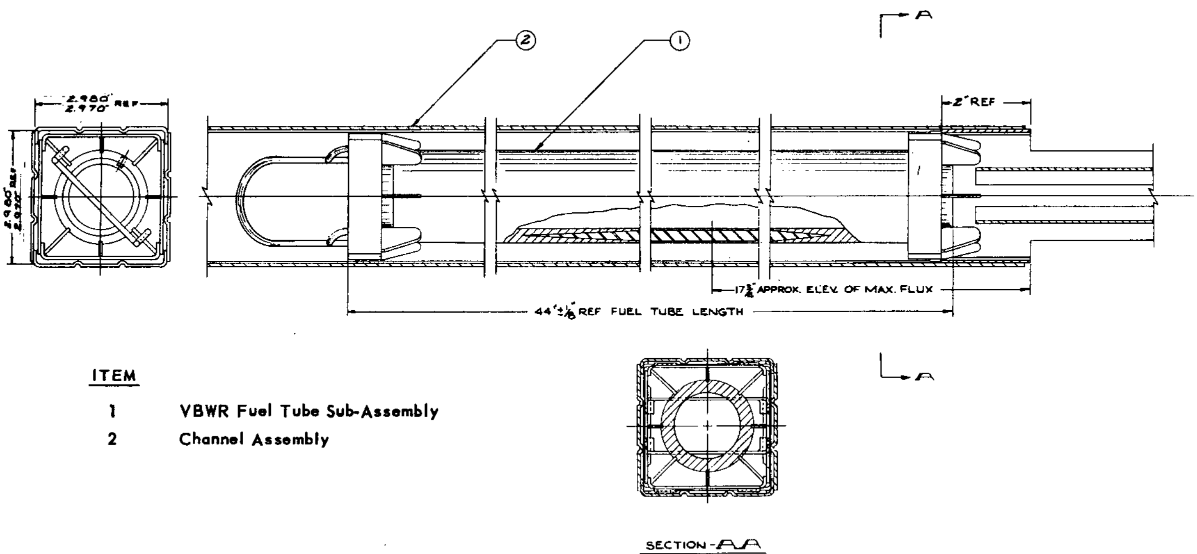

FIG. 2 ASSEMBLY OF U – 2% Zr ELEMENT IN VBWR CHANNEL BOX

TABLE II

MAXIMUM AND AVERAGE IRRADIATION CONDITIONS

Output of element, kw	250	200
Average specific power, MW/T	18.2	14.6
Peak-to-average flux	1.16	1.16
Maximum heat flux, pcu/(ft^2)(hr)		
Inner surface	292,000	233,000
Outer surface	254,000	202,000
Mean coolant temperature, ^{o}C	285	285
Maximum surface temperature, ^{o}C	300	299
Maximum cladding-core interface temperature, ^{o}C	345	335
Maximum core temperature, ^{o}C	433	405
Reactor pressure, psia	1000	1000
Exit steam quality, %		
Inner surface	4	3
Outer surface	2	1½
Test channel inlet subcooling, Btu/lb	10	10

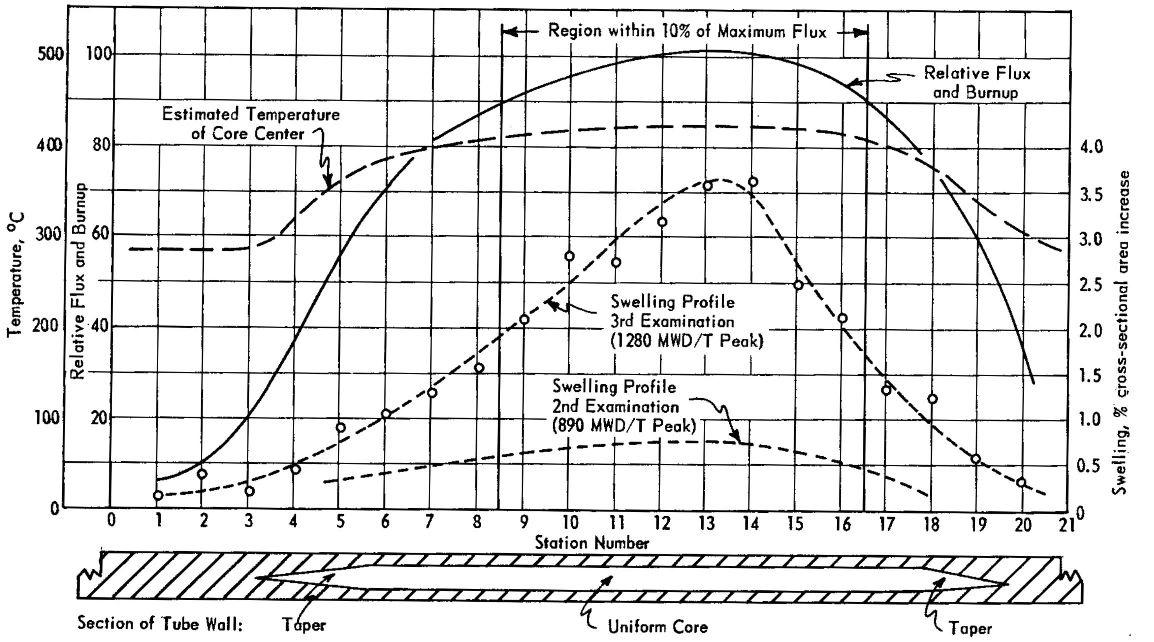

FIG. 3 SWELLING PROFILES OF U – 2% Zr ELEMENT AT 890 AND 1280 MWD/T BURNUP

The element underwent approximately 65 power cycles involving element power changes of 125 kw or more during periods of significant power operation. During the last several months of the U - 2% Zr element exposure, the reactor was run below normal power levels during a number of control rod tests in which the element power varied from 0 to 10 kw and the reactor coolant temperature and pressure varied in the ranges 265 to 285°C and 700 to 1000 psig. There were also frequent reactor cool-downs during this period. The reactor was shut down for a prolonged period following these tests without resuming normal operation at significant power levels.

UNDERWATER EXAMINATION TECHNIQUES

At various intervals during the irradiation, the element was removed from the reactor for examination and a number of measurements were taken underwater for comparison with similar preirradiation measurements. These measurements included outside diameter, inside diameter, bow, length, underwater weight, and a gamma scan. From these measurements, the swelling of the element at various points along its length was established quite accurately, and a correlation of swelling with core temperature and burnup was facilitated. The techniques employed for the various measurements are described briefly below.

OUTSIDE DIAMETER MEASUREMENT

A remote-reading underwater gauge developed by the Savannah River Laboratory (SRL) was employed for all outside diameter measurements.[9]

The measuring head of the instrument has two diametrically opposed probes each with an extension serving as the moving core of a differential transformer. The probe positions are detected in the remote receiving instrument by a null balance system, and the measured diameter is read out directly on a digital indicator.

During each examination of the fuel element, outside diameters were measured at orientations of 0^o, 45^o, 90^o, and 135^o at twenty stations spaced every two inches along the tube length, constituting a total of eighty measurements. Additional measurements were made at the twenty stations at 225^o representing the same diameters as those at 45^o (but with the gauge approaching from the opposite side of the tube) for the purpose of determining probable error in the measurements. The differences between the 225^o and 45^o readings during the various examinations indicated a consistent three-sigma deviation of the order of 1 mil for any one OD measurement, 1/4 mil for the average of twenty measurements, and about 0.1 mil for the average of eighty measurements. These reproducibility results reflect the fact that each reading was made only to the nearest mil. It is believed that the individual readings could actually be read significantly to perhaps 1/4 mil.

The SRL remote-reading gauge proved to be a quick and accurate means of obtaining comprehensive data on the performance of the element at intervals during the course of the irradiation.

INSIDE DIAMETER MEASUREMENT

A second remote-reading gauge was developed by SRL for underwater measurement of inside diameters of the element.[9] The instrument works on the same principle as the outside diameter gauge, but has only one movable probe and differential transformer and a diametrically opposed fixed probe in the measuring head. The inside diameter is read out on a digital indicator that is housed in the same console as the outside diameter indicator.

Inside diameters of the element were measured at the same stations as the outside diameters at orientations of 0^o and 90^o with reproducibility checks at 180^oC. Also, the maximum and minimum diameters were obtained by rotation of the probe through the full tube circumference to determine eccentricity. Comparison of the 0^o and 180^o readings of the various examinations indicates a probable error about twice that of the outside diameter measurements. Failure of the probes to be completely self-aligning on the true diameter because of surface irregularities probably contributed significantly to the inside diameter measurement error. The individual inside diameter changes obtained by subtracting each reading of one examination from the corresponding reading of another are of little significance in most cases encountered because the inside diameter changes were generally not much greater than the probable error involved. The averages of the inside diameter and eccentricity measurements, however, are statistically significant.

LENGTH MEASUREMENT

An aluminum caliper rule was built for measuring the element length to the nearest hundredth of an inch. The device is of sufficient length so that the scale can be read conveniently well above the water surface while the measuring end is inserted through the fuel element under 15 feet of water. Four measurements of element length were made at 90° intervals during each examination with deviation among the four readings generally being 0.01 inch or less.

MEASUREMENT OF BOW

A series of parallel lines, which were ruled on the element horizontal measurement support plate and with which the element profile was visually compared, served as a measure of element bow. The ruled lines were 1/16 inch in width and in spacing so that a deviation in the element profile of 1/16 inch from a straight line could be readily perceived. Observations were made at 45° intervals while the element was in position for the respective outside diameter measurements.

MEASUREMENT OF TUBE VOLUME BY DISPLACEMENT

With the purpose of providing a check on over-all element volume change, the element was weighed in air and in water before irradiation to determine the initial displaced volume; the tube was weighed in water again during each subsequent examination to determine volume change. The weighings were made to the nearest gram by suspension on a wire from an Ohaus platform balance of 20-kilogram capacity. A duplicate element was weighed to the nearest gram in water on three different occasions to establish the weighing accuracy, and the same reading was obtained on each occasion. The accumulation of a crud layer on the element surface became a major factor of uncertainty in the weighing of the irradiated element. However, the volume as determined from dimensional changes was considered to be accurate within several hundredths of a per cent and was not apparently affected by the crud layer. Hence, the immersed weight change of the element, adjusted for the volume change determined dimensionally, became a significant indication of the amount of crud adhering to the element.

RESULTS

The swelling measurements described above were made during examinations of the element at 150, 890, and 1280 MWD/T burnup. The changes observed during these examinations over the values of the preirradiation examination are tabulated in Tables III, IV, and V. The tables are explained below, following which correlations are made of total swelling and swelling per unit burnup versus burnup.

EXPLANATION OF TABULATED DATA

The station numbers in the far left column of each table refer to measuring stations, the locations of which with respect to the element length are shown in Figure 3. The left portion of each table shows the increase in OD at each of eighty points on the tube, as determined by subtracting the preirradiation readings from the current readings. At 150 MWD/T burnup (Table III), only the over-all average outside diameter change (0.10 mils) is significant. At 890 MWD/T (Table IV) the average change for each station is significant, and at 1280 MWD/T (Table V) the swelling has proceeded to the point that the individual changes are quite significant.

The inside diameter changes, which are presented in the central portion of each table, are generally not individually significant even at 1280 MWD/T burnup, although the averages for each station are of the same order as their probable error at the higher burnups. The average inside diameter changes at 0^{o} and 90^{o} differ quite significantly at the higher burnup, and a significant change is observed in over-all eccentricity.

Eccentricity was determined by subtracting the shortest diameter from the longest diameter observed in rotating the inside diameter probe through 180^{o}. The eccentricity thus determined averaged 4.5 mils over the length of the tube at 1280 MWD/T compared to a preirradiation value of 4.1 and a 890 MWD/T value of 5.2. The longer diameters were predominantly in the 0^{o} plane at 1280 MWD/T but not to the extent observed in previous readings. Commensurately, the inside diameters measured in the 0^{o} plane have decreased considerably more than those of the 90^{o} plane at 1280 MWD/T whereas at 890 MWD/T the decrease was chiefly in the 90^{o} plane. Since the outside diameters measured in the 0^{o} plane and 90^{o} plane have grown at essentially the same rate, it appears that there is no significant change in real eccentricity but rather that the tube wall swelled preferentially and "inwardly" in the 0^{o} plane during the latter part of the test compared to a preference for the 90^{o} plane up through 890 MWD/T. This effect might be related to some extent to the tube orientation in the reactor. During the first 890 MWD/T burnup the 90^{o} radius of the element was oriented toward the high flux region of the reactor where the flux is estimated to have been about 5% higher than the average. Orientation during the remainder of the test was not determined.

No significant change in length or bow of the element was observed during any of the examinations. The per cent cross-sectional area increase, calculated from the average outside and inside diameter changes for each station, represents the local volume change, assuming no local length changes. These values are plotted in Figure 3 for the 890 and 1280 MWD/T examinations, along with the relative burnup curve obtained from a longitudinal gamma scan of the element at 150 MWD/T.

The over-all tube volumetric growth calculated on the above basis was about 1.52% at 1280 MWD/T burnup. This value is believed to be accurate within several hundredths of a per cent. Calculation of the same value from immersed weighings before and after irradiation indicates a volume increase, ignoring the effect of crud, of 1.25 ±0.05%.

TABLE III

COMPARISON OF 150 MWD/T AND PREIRRADIATION MEASUREMENTS

Station No.	Increase in OD, mils					Decrease in ID, mils			Eccentricity Increase, mils	Cross-sectional Area Increase, %	Relative Flux (from Gamma Scan)
	0°	45°	90°	135°	Avg	0°	90°	Avg			
1	0	1	0	-1	0.00	-2	3	0.5	8	0.07	6
2	1	0	0	0	0.25	0	-1	-0.5	-1	-0.02	11
3	2	0	1	0	0.75	0	0	0.0	2	0.15	21
4	0	0	0	0	0.00	-1	0	-0.5	-1	-0.07	41
5	0	0	1	-1	0.00	1	4	2.5	-1	0.35	59
6	1	0	0	0	0.25	1	1	1.0	-2	0.19	73
7	1	1	0	0	0.50	0	1	0.5	1	0.17	82
8	0	0	0	0	0.00	0	1	0.5	-1	0.07	87
9	1	0	1	0	0.50	0	1	0.5	0	0.17	92
10	1	0	0	1	0.50	0	2	1.0	5	0.24	97
11	-1	-1	0	1	-0.25	0	0	0.0	0	-0.05	99
12	0	0	0	1	0.25	0	0	0.0	1	0.05	100
13	0	0	0	0	0.00	0	2	1.0	1	0.14	100
14	-1	0	-1	0	-0.50	0	1	0.5	0	-0.03	100
15	0	-1	0	0	-0.25	0	0	0.0	0	-0.05	97
16	0	1	-1	0	0.00	0	2	1.0	1	0.14	93
17	0	0	1	1	0.00	-1	0	-0.5	1	-0.07	85
18	-1	0	1	0	0.00	1	0	0.5	-2	0.07	73
19	-1	0	0	0	-0.25	2	3	2.5	0	0.30	55
20	1	0	0	0	0.25	0	3	1.5	0	0.26	33
Avg	0.20	0.05	0.15	0.00	0.10	0.1	1.1	0.6	0.60	0.104	

	0°	90°	180°	270°	Avg
Length increase, inch	0.00	0.00	+0.01	-0.01	0.00

	0°	45°	90°	135°	180°
Bow (to 1/16 inch), inch	0.0	0.0	0.0	0.0	0.0

Appearance: Thin reddish-brown film over essentially all surfaces

	Preirradiation	Postirradiation
Volume by displacement		
Dry weight, grams	16,162	(16,162)
Immersed weight, grams	14,959	14,961
Displaced water, grams	1,203	1,201
Pool temperature, °C	34	35
Water density	0.99440	0.99406
Tube volume, ml	1,210	1,208
Calculated immersed weight of crud, grams	-	3.3

TABLE IV

COMPARISON OF 890 MWD/T AND PREIRRADIATION MEASUREMENTS

Station No.	Increase in OD, mils 0°	45°	90°	135°	Avg	Decrease in ID, mils 0°	90°	Avg	Eccentricity Increase, mils	Cross-sectional Area Increase, %	Relative Flux (from 150 MWD/T Gamma Scan)
1	0	0	1	0	0.25	-1	0	-0.5	5	-0.03	6
2	1	1	1	0	0.75	1	-1	0.0	-4	0.17	11
3	2	0	2	0	1.00	1	0	0.5	0	0.30	21
4	0	0	1	1	0.50	-2	0	-1.0	1	-0.05	41
5	1	0	2	2	1.25	3	7	5.0	2	1.07	59
6	1	2	0	1	1.00	2	1	1.5	-1	0.46	73
7	2	1	1	1	1.25	1	1	1.0	2	0.43	82
8	1	1	1	3	1.50	1	1	1.0	0	0.49	87
9	3	0	2	3	2.00	0	2	1.0	2	0.59	92
10	3	2	2	2	2.25	3	2	2.5	2	0.89	97
11	2	0	2	3	1.75	2	1	1.5	2	0.62	99
12	2	1	2	4	2.25	1	2	1.5	2	0.73	100
13	2	2	2	3	2.25	1	2	1.5	1	0.73	100
14	2	2	1	3	2.00	1	3	2.0	0	0.75	100
15	2	0	2	3	1.75	0	2	1.0	2	0.54	97
16	2	1	1	2	1.50	0	2	1.0	1	0.49	93
17	2	1	1	1	1.25	-2	1	-0.5	1	0.19	85
18	0	0	3	2	1.25	0	0	0.0	-1	0.27	73
19	-1	-1	1	1	0.00	0	4	2.0	2	0.32	55
20	1	-1	1	1	0.50	-3	6	1.5	3	0.35	33
Avg	1.40	0.60	1.45	1.80	1.32	0.45	1.80	1.13	1.1	0.47	

	0°	90°	180°	270°
Length increase, inch	0.00	0.00	0.01	-0.01

	0°	45°	90°	135°	180°
Bow (to 1/16 inch), inch	0.0	0.0	0.0	0.0	0.0

Appearance: Thin rust-colored deposit over central portion of element

	Preirradiation	Postirradiation
Volume by displacement		
Dry weight, grams	16,162	(16,162)
Immersed weight, grams	14,959	14,961
Displaced water, grams	1,203	1,201
Pool temperature, °C	34	38
Water density	0.99440	0.99299
Tube volume, ml	1,210	1,210
Calculated immersed weight of crud, grams	-	5.0

TABLE V

COMPARISON OF 1280 MWD/T AND PREIRRADIATION MEASUREMENTS

Station No.	Increase in OD, mils 0°	45°	90°	135°	Avg	Decrease in ID, mils 0°	90°	Avg	Eccentricity Increase, mils	Cross-sectional Area Increase, %	Relative Flux (from 150 MWD/T Gamma Scan)
1	-1	1	2	0	0.50	1	-1	0	5	0.11	6
2	1	1	2	1	1.25	1	0	0.50	-3	0.35	11
3	1	0	2	0	0.75	0	0	0	0	0.16	21
4	2	1	2	2	1.75	-1	1	0	0	0.38	41
5	3	1	2	3	2.25	4	1	2.50	-1	0.89	59
6	3	2	2	3	2.50	4	2	3.00	-1	1.02	73
7	5	4	5	4	4.50	4	0	2.00	0	1.29	82
8	6	5	5	5	5.25	2	3	2.50	-1	1.54	87
9	9	6	8	8	7.75	3	2	2.50	-1	2.08	92
10	11	9	11	11	10.50	4	2	3.00	0	2.76	97
11	13	8	13	15	12.25	1	0	0.50	1	2.74	99
12	14	11	13	16	13.50	2	1	1.50	6	3.16	100
13	15	12	14	15	14.00	5	2	3.50	-1	3.59	100
14	15	12	12	16	13.75	7	1	4.00	2	3.62	100
15	14	8	10	14	11.50	1	-1	0	1	2.49	97
16	10	7	7	9	8.25	2	2	2.00	0	2.11	93
17	5	4	5	4	4.50	5	-1	2.00	0	1.30	85
18	2	2	4	3	2.75	5	3	4.00	0	1.24	73
19	1	2	3	2	2.00	2	0	1.00	1	0.59	55
20	0	-1	1	0	0.00	8	-4	2.00	1	0.32	33
Avg	6.45	4.75	6.15	6.55	5.98	3.00	0.65	1.83	0.45	1.59	

	0°	90°	180°	270°	Avg
Length increase, inch	0.03	0.04	0.05	0.02	0.035

	0°	45°	90°	135°	180°
Bow (to 1/16 inch), inch	0.0	0.0	0.0	0.0	0.0

Appearance: Thin rust-colored deposit over hottest two-thirds of element estimated to be a fraction of a mil thick

	Preirradiation	Postirradiation
Volume by displacement		
Dry weight, grams	16,162	(16,162)
Immersed weight, grams	14,959	14,939
Displaced water, grams	1,203	1,223
Pool temperature, °C	34	18.5
Water density	0.99440	0.99850
Tube volume, ml	1,210	1,225
Calculated immersed weight of crud, grams	-	3.3

The difference in these two values can be interpreted as a buildup of surface crud having an immersed weight of 3.3 grams. A similar calculation at 890 MWD/T indicated the presence of 5.0 grams of crud. Assuming a crud density in the range two to five grams per cubic centimeter, the average thickness of the crud over this area is calculated to be between 0.1 and 0.5 mils. Observations of the tube surface by periscope, especially where the outer diameter probes had scored through the crud revealing the tube surface, resulted in crud thickness estimates of this same order.

CORRELATION OF RESULTS

It is apparent from both the shape and height of the 1280 MWD/T swelling profile in Figure 3 compared with the 890 MWD/T profile that the amount of swelling increased sharply between these burnup levels. In the 16-inch length of tube exposed to at least 90% of peak flux, the average cross-sectional growth at the 890 MWD/T examination was 0.67% and the average burnup 865 MWD/T; the swelling in this region was equivalent to a 6.7% volume increase per atom per cent burnup. In the same region at the 1280 MWD/T examination, the average cross-sectional growth was 2.82% and the average burnup 1235 MWD/T, corresponding to a 19.6% volume increase per atom per cent burnup or three times that at the lower burnup. Comparison of swelling and burnup in various sections along the length of the tube reveals a continuous but nonlinear relationship, the rate of growth increasing with burnup. Swelling is plotted as a function of burnup in Figure 4; the points of the curve are taken from all three examinations of the element. All points used represented measurements taken over the uniform region of the element core. The estimated maximum temperature at which that portion of the core operated is given on the plot for each point. Temperature does not appear to be a significant variable in the narrow temperature range involved, in view of the relatively smooth transition from the 890 MWD/T high temperature points to the 1280 MWD/T low temperature points in Figure 4.

The swelling per unit burnup, rather than the total swelling, is plotted against burnup in Figure 5. Through the first 700 MWD/T burnup, the element apparently swelled at about the theoretical rate corresponding to the differential volume of the new atoms being generated. Beyond 700 MWD/T, the swelling rate increased sharply through the remainder of the exposure, probably as a result of the fission gas expanding as a gas upon agglomerating in minute bubbles in the metal core.

At 1280 MWD/T, when the irradiation was discontinued because of the VBWR being shut down for a prolonged period, the peak burnup region of the element had swelled 3.6% in volume; the swelling was manifested largely in a 14-mil increase in outside diameter. This diameter increase is equivalent to a strain in the outer cladding of 0.7%, which is about 20 or 30% of the strain considered tolerable in the Zircaloy cladding of this type of element.

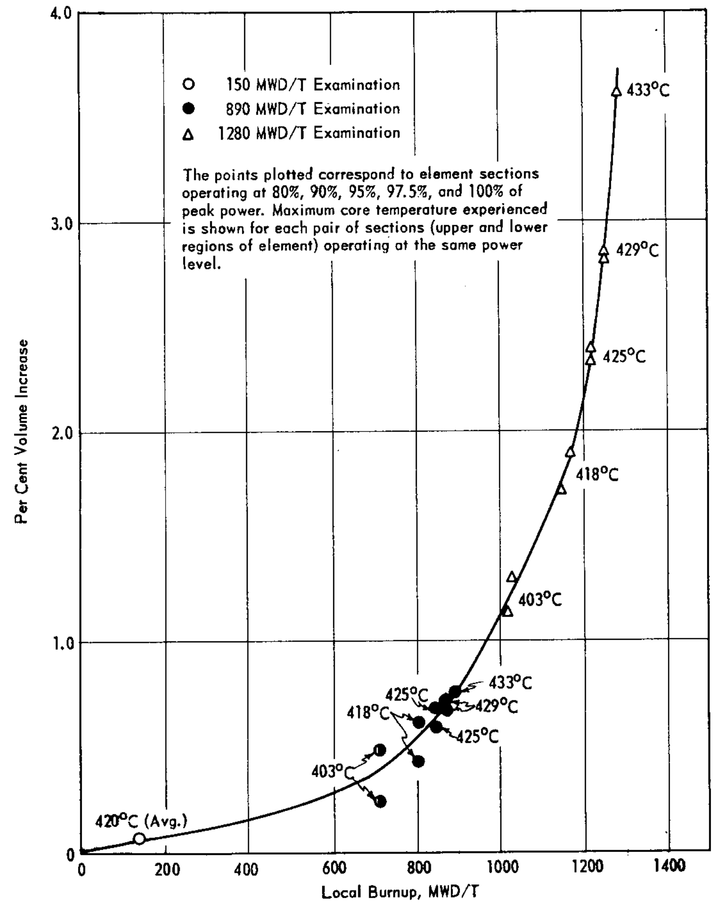

FIG. 4 SWELLING OF U – 2% Zr ELEMENT VS. BURNUP

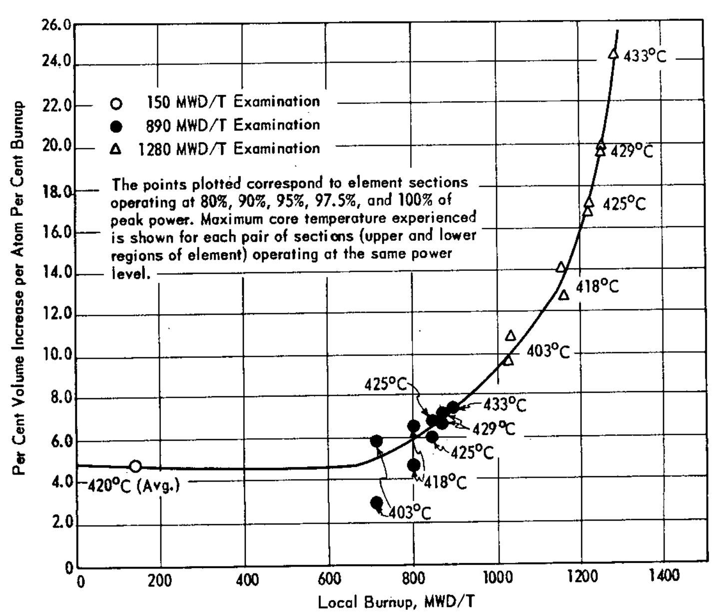

FIG. 5 SWELLING PER UNIT BURNUP VS. BURNUP

BIBLIOGRAPHY

1. Heavy Water Moderated Power Reactors - Progress Reports. Compiled by R. R. Hood. E. I. du Pont de Nemours & Co., Tech. Div., Wilmington, Del. AEC Research and Development Reports DP-232, 245, 265, 285, 295, 315, 345, 375, 385, 395, 405, 415, 425, 435, 445, 455, 465, 475, 485, 495, 505, 515, 525, 535, 545, 555, 565, 575, 585, 595, 605, 615, 625, 635, 645, 655, 665, 675, 685, 695, 705. (July 1957 - December 1961).

2. Isakoff, L. Economic Potential for D_2O Power Reactors. E. I. du Pont de Nemours & Co., Technical Division, Wilmington, Del. AEC Research and Development Report DP-570, 76 pp. (February 1961).

3. Heavy Water Moderated Power Reactors - Progress Reports - August, September, and October 1957 and November 1957 through January 1958. Compiled by L. Isakoff. E. I. du Pont de Nemours & Co., Tech. Div., Wilmington, Del. AEC Research and Development Reports DP-265 (February 1958) and DP-285 (March 1958).

4. Kaufmann, A. R., et al. "Zirconium Cladding of Uranium and Uranium Alloys by Coextrusion". pp. 157-81 of Fuel Elements Conference, Paris, November 18-23, 1957. Commissariat à l'Énergie Atomique, Paris, France and U. S. Atomic Energy Commission, Washington, D. C. AEC Research and Development Report TID-7546 (Book 1), 371 pp. (March 1958).

5. Pugh, R. C. Irradiation of a Uranium Metal Fuel Tube at Chalk River. E. I. du Pont de Nemours & Co., Wilmington, Del. AEC Research and Development Report DP-667, 12 pp. (November 1961).

6. Kaufmann, D. F. and W. J. Richmond. Evaluation of Six Zircaloy-Clad U - 2 w/o Zr Alloy Tubes Made for Irradiation in the VBWR Test Loop. Nuclear Metals, Inc., Concord, Mass. AEC Research and Development Report NMI-4387 (del.) 66 pp. (June 1959).

7. "Reactor File No. 4 - Vallecitos Boiling Water Reactor", Nucleonics 16, No. 2 (February 1958).

8. Kornblith, L., S. Untermyer, L. Welsh and W. H. Nutting. Vallecitos Boiling Water Reactor. General Electric Co., San Jose, Calif. Report No. GER-1494 (Reprint of paper presented at the 1958 Nuclear Congress, Chicago, Ill., March 18, 1958).

9. Goodwin, L. E. A Differential Transformer Gauge for Outside and Inside Diameters of Long Tubes. E. I. du Pont de Nemours & Co., Wilmington, Del. AEC Research and Development Report DP-263, 15 pp. (January 1958).