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FINAL REPORT: IRRADIATION PERFORMANCE OF
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PT-IP-123-A.

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**FINAL REPORT: IRRADIATION PERFORMANCE OF ENRICHED
URANIUM CLAD IN STAINLESS STEEL PT-IP-123-A.**

By T. T. Claudson

June 9, 1959

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FINAL REPORT: IRRADIATION PERFORMANCE OF ENRICHED
URANIUM CLAD IN STAINLESS STEEL. PT-IP-123-A.

I. INTRODUCTION

Early in the development of an extended surface fuel element for use in the NRP, several 7-rod cluster fuel elements were irradiated to determine the dimensional stability of such geometries at high burnups. These elements were fabricated from small diameter uranium rods clad unbonded in stainless steel tubes and assembled in a rod cluster geometry by various support devices. Zircaloy clad fuel rods were not yet available, the stainless steel clad rods therefore served as a suitable material which would withstand high temperature water over a long period of time and maintain relatively high strength properties.

The purpose of the irradiation was to determine the effect of high exposure on the swelling, dimensional stability, microstructure, and physical properties of uranium rods restrained unbonded in stainless steel. At the same time, this test was designed to, (1) evaluate the effect of fuel rods operating in a cluster geometry, (2) to monitor the central core temperature of the uranium, (3) to determine the stainless steel-uranium interface heat transfer bond coefficient, and (4) to determine the average specific power of the assembled element. Goal exposure for this irradiation test was 3500 MWD/T.

II. SUMMARY

Four 7-rod cluster fuel elements operated successfully to an exposure of 0.41 a/o burnup (3250 MWD/T) in the K Through hole irradiation facility. During the test, data were collected making it possible to calculate the fuel specific power and the heat transfer bond coefficient between the uranium and stainless steel clad. These values at the beginning of the test were 119 kw/ft and 1350 BTU/hr-ft²-°F respectively. After 270 days of operation, the specific power was 75 kw/ft and the bond coefficient had increased to about 3000 BTU/hr-ft²-°F. The measured uranium core temperature dropped from 530°C to 280°C during the 270 days of operation.

Post irradiation examination showed that the uranium volume had increased about 2.5% filling all available space within the cladding. The density had decreased to 18.68 grams/cc and the hardness increased to R_A 72. No uranium grain structure was evident. Electron microscopy revealed that swelling voids were present and that some uranium stainless steel bonding did exist in the end cap region. There was no evidence of impending fuel failure.

III. DESCRIPTION OF TEST ASSEMBLY

Each of the four fuel elements used in the test consisted of 7 uranium rods enriched to 1.35 a/o U-235, 0.499±.001 inch in diameter and 8 inches in length. These rods were jacketed unbonded in 0.030 inch thick, 304L stainless steel tubing, 0.560 inch in outside diameter. Stainless steel end caps were welded in place by the heliarc welding process and the entire assembly held together by an external ring and clip support. This assembly and fixtures used to assemble the elements are shown in Figure 1. Two end caps were specially designed to admit thermocouple leads. These chromel-alumel

FIGURE 1

7 rod cluster fuel element assembly showing component parts and fixtures.

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thermocouples, magnesia insulated and stainless steel clad, were welded to the end cap by the vacuum electron beam welding technique. This effected a gas-tight, water-tight seal, protecting the thermocouples and providing a means of measuring the central core temperature of the fuel rods. Two opposite peripheral rods were chosen as the monitor rods as a means of checking the accuracy of the thermocouple readings. The fuel element was placed downstream of the other three fuel elements in order to measure the maximum possible core temperature.

In order to allow for the maximum safety of the KE Through Hole Facility process tube, a basket was fabricated into which were placed the fuel elements. The elements, pinned together, were held securely by end fittings welded to the aluminum basket. These end fittings also held Swagelok fittings through which were placed two thermocouples upstream and two thermocouples downstream of the basket. These thermocouples measured the temperature difference across the basket from which the average specific power of the fuel could be calculated. The six thermocouples were brought out of the reactor through a single conduit tube and connected to recorders located in the X-0 level room.

IV. IN-REACTOR DATA

The temperature across the fuel basket assembly and the central core temperature in two peripheral rods of one cluster element were recorded continuously for 270 days. No indication of thermocouple failure was seen during 187 days of this period. At this time, one thermocouple in a peripheral rod showed erratic readings and data from this thermocouple were henceforth discarded. From the data collected, the specific power of the fuel and the heat transfer bond coefficient between the uranium and its stainless steel jacket was calculated. The heat transfer bond coefficient for the irradiation test can be expressed as:

$$(1) \quad h = 26,170 \frac{Q_r}{\Delta T_I} \quad \text{where}$$

h = heat transfer bond coefficient, BTU/hr-ft²-°F

Q_r = specific power, kw/ft - rod

ΔT_I = temperature difference between the uranium surface and jacket I.D.

The value for ΔT_I can be expressed as

$$\Delta T_I = [T_c - \Delta T_u] - [T_{BW} + \Delta T_J] \quad \text{where}$$

T_c = U core temperature, °F

ΔT_u = ΔT across U core, °F

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$\Delta T_F = \Delta T$ coolant film, $^{\circ}F$

$\Delta T_J = \Delta T$ across jacket, $^{\circ}F$

T_{BW} = bulk water temperature, $^{\circ}F$

Substituting in all known values for this experiment:

$$(2) \Delta T_I = T_c - T_{BW} - 34.1 Qr$$

The specific power of the fuel was calculated from the expression:

$$(3) q = WC_P \Delta T \quad \text{where}$$

Q = total heat generated BTU/hr

C_P = specific heat of coolant BTU/#- $^{\circ}F$

W = rate weight of coolant #/hr

ΔT = temperature difference across fuel basket, $^{\circ}F$

By use of equations 1, 2, and 3 the heat transfer bond coefficient and the fuel specific power was calculated. A summary of this data is plotted and shown in Figure 2. Considerable spread is exhibited by the data points but in general it can be stated that the heat transfer bond coefficient increased during the irradiation, the rate decreasing towards the end of the test. The specific power of the fuel tended to decrease during the test. This plus the increase in bond coefficient lowered the central core temperature of the fuel from 530 C at the beginning of the test to 280 C after 0.41 a/o burnup. It was noticed that during the first few startups of the reactor, the bond coefficient tended to be lower than it was during the initial startup. This was explained by the clad being strained on the initial startup by the thermal expansion of the uranium. Subsequent startups therefore did not apply as much pressure to the clad, thus causing a lower heat transfer bond coefficient. The details of the first week of operation were previously reported. (1)

V. POST-IRRADIATION EXAMINATION DATA

One fuel element was submitted to Radiometallurgy Examination Operation for examination. Figure 3 shows the element as it was received for examination. Due to the split ring type of spider used, it was difficult to handle the element without damaging the fuel supports. Such damage as seen in Figure 3 was caused by handling and was not due to damage incurred during irradiation.

The rods were separated and each examined for warpage, hot spots, excess corrosion and other signs of impending fuel failure. Except for a light film on the jacket surface, the rods looked the same as before irradiation. In order to determine any dimensional changes which might have occurred, the outside diameter and length of each rod was measured. Measurements were made

(1) Claudson, T. T., "Irradiation Data - 1.3% Enriched 7-Rod Cluster Fuel Elements," HW-56233.

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FIGURE 2

Heat transfer bond coefficient between uranium and 304 S.S.
vs. time as a function of specific power.

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FIGURE 3

Photographs showing longitudinal and end views of 7 rod cluster fuel element irradiated to 3250 MWD/T.

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by both micrometer and optical measurement instruments. Since it was impossible to remove a rod from its clad, sections of the clad uranium rod were made and polished for optical measurement of the uranium and metallographic examination. Table I is a tabulation of all pre and post-irradiation data obtained from the examination including density measurements and burnup analysis. From measurements taken, a 0.002 to 0.003 inch decrease in diameter of the uranium was found. The uranium length had also increased an average of 0.032 inch filling the space between the fuel and end cap left for thermal expansion and growth. The uranium length increase measurements were found by subtracting from the overall length of the fuel, the length of each end cap. It was impossible in some cases to remove the end caps due to metallurgical bonding between the uranium core and the end caps.

TABLE I

Rod No.	Uranium Diameter		Uranium Length	
	Pre	Post	Pre	Post
8	0.4985	0.502	8.0025	8.041
9	0.4985	0.500	8.0015	8.030
10	0.4985	0.502	8.0015	8.034
11	0.4980	0.502	8.0015	8.029
12	0.4985	0.501	8.0005	8.031
13	0.4980	0.500	8.0015	8.031
14 (center rod)	0.4980	0.501	8.0010	8.042

BURNUP ANALYSIS

Rod No.	a/o Burnup
11	0.414 a/o
14	0.327 a/o

DENSITY MEASUREMENT

Rod No.	Density	
	Pre	Post
11	18.875	18.685
11		18.704
11		18.700
11		18.695
14		18.763

Both transverse and longitudinal sections were prepared for metallographic examination and replication for electron microscopy. Sections from one peripheral rod and the center rod were also prepared for density and hardness measurements and for burnup analysis. After polishing and cathodically etching, examination of the rods showed complete lack of grain structure throughout the uranium

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cross section. Macrocracks were also seen across the uranium parallel to one another. It was determined however, that these cracks were formed by the cutting wheel while preparing samples for examination and were not the result of the irradiation. No further cracking was found in any of the samples observed.

Before the irradiation there existed a 0.001 to 0.002 inch clearance between the uranium and cladding. After irradiation, no clearance was found and close examination showed that the uranium had formed a very tight mechanical bond by flowing into small crevices in the inside surface of the cladding. Figures 4 and 5 are photomicrographs of the central and edge areas of the uranium respectively.

As previously mentioned, the uranium rods had increased in length until contact was made with the end caps. Close examinations revealed that not only had the uranium made contact with the end cap, but it had also formed a metallurgical bond. Work by T. K. Bierlein and B. Mastel through electron microscopy revealed at high magnification small areas of a uranium-stainless steel bonding along the interface. They also discovered small swelling bubbles throughout the uranium structure. Further study by optical microscope revealed areas of uranium between the end cap and cladding extending up to the weld. An increase in diameter of 0.012 inch along this portion of the rod was the result of such uranium creep. Figure 6 shows the uranium-end cap interface including the end cap-cladding area. Figure 7 shows the uranium as it formed around the end cap and between the cladding.

Burnup analysis were made on one peripheral and one center rod of a cluster. From Table I the peripheral rod analysis showed a burnup of 0.414 a/o compared to 0.312 a/o burnup on the center rod. Results from several cluster irradiations have shown the same results in that the center rod receives about 80% of the exposure gained by the peripheral rods. Density of the center rod was higher than in the outer rod, indicating a lower burnup.

VI. CONCLUSION

This irradiation test represents one of the most successful instrumented tests ever run at HAPC. Previous estimates of the uranium stainless steel heat transfer coefficient have been as high as 6000 BTU/hr-ft²-°F. Actually, this value has now been determined to be about half this value and should lead to more accurate knowledge of temperature distributions in such unbonded fuel elements. Although NPR fuel elements will be clad-bonded to Zircaloy-2, it is feasible that fuel elements for future reactors will be clad in stainless steel especially where reactor operation calls for exposures up to 10,000 MWD/T.

Acknowledgements are given to R. K. Marshall, Irradiation Testing Operation for the assembly of the instrumentation and the charging and collection of data during the entire run of the test. Also to R. Teats, Radiometallurgy Examination Operation, for the thorough post-irradiation examination of the fuel.

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FUEL ELEMENT DESIGN OPERATION
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Photomicrograph of uranium after 0.41 a/o burnup and operating at a measured core temperature of 530-280 C. 100X cathodic etch.

FIGURE 4

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FIGURE 5

Photomicrograph of uranium-stainless steel interface after 0.41 a/o burnup and operating at a measured core temperature of 530-280 C. 300 X cathodic etch.

FIGURE 6

Photomicrograph showing uranium-end cap region after 0.41 a/o burnup. Maximum core temperature 530C. Cathodic etch. 50X.

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S.S. End Cap

S.S.
Clad

Uranium

FIGURE 7

Photomicrograph of 1.35% enriched uranium flowing around stainless steel end cap. Cathodic etch. 300X.

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