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Radiation Effects on
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EFFECTS OF IRRADIATION ON CAST SPECIMENS
OF URANIUM-CHROMIUM EUTECTIC ALLOY

by

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EFFECTS OF IRRADIATION ON CAST SPECIMENS OF URANIUM-CHROMIUM EUTECTIC ALLOY

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ABSTRACT

In order to determine the irradiation behavior of uranium-chromium eutectic alloy (U-5 w/o Cr), which is being considered as fuel material in the proposed Dow-Detroit Edison reactor, cast enriched specimens of the alloy were irradiated in the MTR in NaK-filled capsules to total atom burnups ranging up to 0.65% (5500 MWD/T). The specimens were made at Battelle Memorial Institute. The alloy was irradiated both bare and clad with zirconium, stainless steel, or Globe iron. Although none of the specimens showed large length changes resulting from irradiation, they all developed extreme surface roughening, which could be only partially restrained by the claddings investigated. The general deterioration of the specimens under irradiation apparently resulted from an undesirably large grain size, and it was recommended that future irradiations on the alloy be made only on specimens known in advance to be fine-grained.

INTRODUCTION

As visualized in late 1953, the core of the proposed reactor to be constructed by the Dow-Detroit Edison Nuclear Power Project (now the Atomic Power Development Associates) was to be an essentially cylindrical shape made up of cast segments of uranium-chromium eutectic (U-5 w/o Cr) alloy. Each segment, which was in the form of a thin-webbed hexagon, was to be pierced by a large number of tubes through which sodium was to be circulated for heat removal. The eutectic uranium-chromium alloy appeared attractive for application in the reactor because of its reportedly excellent castability. This property would be of advantage not only in the production of the initial loading of hexagonal segments but in subsequent metallurgical reprocessing of the irradiated fuel. No information was available on the irradiation behavior of this alloy and, accordingly, representatives of the Dow-Detroit Edison group began investigating various means by which the irradiation characteristics of the alloy could be determined.

In 1952 the Metallurgy Division of Argonne National Laboratory had begun an investigation of the effects of irradiation on a number of fuel materials. The MTR was being used as an irradiation facility and the

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experimental irradiation specimens were contained in capsules in which NaK was used to transfer heat from the specimen to the capsule wall where it was then removed by reactor cooling water.⁽¹⁾ This arrangement appeared to be the most suitable for the contemplated U-5 w/o Cr alloy irradiations. Arrangements were therefore made between Dow-Detroit Edison and Argonne personnel for the proposed irradiations, which were to be done as a part of the general study being made by Argonne on the effects of irradiation on fuel materials. As originally outlined, the irradiations were to include exposures both in the MTR, for burnup by thermal neutrons, and in the EBR, for burnup by fast neutrons.

DESCRIPTION OF SPECIMENS AND IRRADIATION PROCEDURE

Both bare and clad specimens in the form of short tubes were made for irradiation studies in the MTR. The dimensions of the fuel section in each specimen were 0.55 in. OD x 0.05 in. annular thickness and 1.00 in. in length. In order to achieve the desired temperatures and burnup rates, the alloy was enriched so that the composition was 79.2 w/o U²³⁸, 15.8 w/o U²³⁵, 5 w/o Cr. The selected claddings were Globe iron, zirconium, and stainless steel. Over-all dimensions of the clad specimens were 0.38 in. ID, 0.52 in. OD, 1.375 in. length, fuel alloy length 1.00 in., and 0.010 in. cladding thickness.

The specimens were fabricated at Battelle Memorial Institute. Bare specimens were reportedly cast at 1900°F (1038°C) into a graphite mold at 1100°F (593°C), while clad specimens were cast at 2000°F (1093°C) into a graphite mold at 1200°F (649°C).⁽²⁾ The composition of the alloy as stated on the transfer papers was: 77.8 w/o U²³⁸, 17.2 w/o U²³⁵, 5.0 w/o Cr. Seven bare, seven Globe iron, six zirconium, and seven stainless steel clad specimens were supplied to Argonne for the MTR irradiations. In addition, sufficient unenriched specimens, both bare and zirconium-clad, were supplied to make up two EBR fuel rods. The specimens were solid cylinders with over-all dimensions of 0.364 in. diameter and 1.5 in. length.

Five MTR specimens of each type were placed in zirconium capsules with 22 ml NaK contained in each capsule for heat transfer. In order to center each specimen in its capsule and to permit convection in the NaK a centering spacer was used. A typical bare specimen is shown in Figure 1 and the same specimen is shown in Figure 2 with centering spacers in place.

Each capsule contained a small piece of monitor alloy (99.0 w/o Al, 0.50 w/o Co, 0.49 w/o Mn) which could be gamma counted by a calibrated procedure to obtain the thermal flux incident on the capsule. Because of the enrichment used, however, there was expected to be an appreciable but unknown amount of self-shielding in each specimen, so that atom burnups in

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the specimens or irradiation temperatures could not be computed accurately. Calculations made by W. J. McCarthy of the Dow-Detroit Edison group gave maximum specimen temperatures in a flux of 1×10^{13} to be 403°C for bare specimens and 622°C for clad specimens.

RESULTS AND DISCUSSION

The first group of MTR-irradiated specimens was examined in August, 1954, and consisted of two bare specimens and one Globe iron, one zirconium, and one stainless steel clad specimen. The irradiation conditions and results of post-irradiation measurements for each specimen are given in Table I. Thermal cycling of the U-Cr cast eutectic has yielded qualitatively similar results (see Appendix I).

The length changes shown in Table I are relatively small when compared with other fuel materials that have been exposed to similar atom burnups. On the other hand, surface stability of the specimens was exceptionally poor, as shown in Figures 3 through 7. Both bare specimens, shown in Figures 3 and 4, were so distorted that the centering spacers could not be removed. The bare specimen with the higher burn-up, shown in Figure 4, developed a crack along about a third of its length.

Of the clad specimens, stainless steel appeared to be most effective in restraining the surface roughening of the core alloy, as shown in Figure 6, whereas zirconium was least effective, as shown in Figure 5.

As this first group showed conclusively that the entire 20 MTR specimens as supplied were quite unsatisfactory as possible fuel material, no examinations were made of additional irradiated specimens. No examination has yet been made of the EBR specimens, which are at present still under irradiation in that reactor. However, they were fabricated in the same manner as the MTR specimens and therefore can be expected to show the same irradiation effects.

Based on previous results from other materials, the general deterioration of these specimens which resulted from the irradiation they received is the result of an undesirably large grain size. The specimens were cast into a heated mold,⁽²⁾ so that a coarse grain size could form even in such thin sections on solidification and cooling. Since the irradiation results were obtained additional studies on the alloy have been reported by Battelle.⁽³⁻⁶⁾ These investigations confirm the presence of large grains in the alloy when it is allowed to cool slowly. They also show that the grain size can be greatly reduced by suitable heat treatment or by slight changes in composition.

Fine grain size is an essential requirement for optimum surface stability in anisotropic fuel alloys under irradiation. Thus, the likelihood of radical distortion can be detected by pre-irradiation metallography.

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Experience has also shown⁽⁷⁾ that the characteristics of surface damage are simulated reasonably well by thermal cycling tests, although these are not always reliable as an indication of length changes to be expected.

CONCLUSIONS

The eutectic uranium-chromium alloy in the condition in which it was supplied to Argonne was unsuitable for use as fuel material because of an undesirably large grain size. Further irradiation exposures of the alloy should await the attaining of a suitably fine grain size. Of the cladding materials investigated, stainless steel was most effective, Globe iron was not quite as suitable, and zirconium showed the least ability to restrain the surface roughening of the core material. The use of metallography or thermal cycling is indicated as a means of screening materials before selection for irradiation.

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Table I

Specimen	Cladding	Thermal nv x 10 ¹³ (a)	Total Atom Burnup, % ^(b)	Length Change, %	Growth Rate, Micro-in./in. ppm Burnup
AD- 1	None	1.76	0.33	3.1	9.4
- 2	None	1.58	0.65	-0.9	-1.4
- 6	zirconium	0.87	0.12	-0.53	-4.4
-11	stainless steel	1.13	0.23	-0.87	-3.8
-16	Globe iron	1.17	0.25	0.52	2.1

(a) Incident flux at upper end of irradiation capsule.

(b) Based upon incident flux at top of capsule. The magnitude of corrections for flux gradients and for flux depression and self-shielding by the specimen are unknown and were therefore not included in the burnup calculations.

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APPENDIX IThermal Cycling of As Cast Uranium-Chromium Eutectic

Mayfield and Chiswik have thermally cycled three nominal 20 a/o as cast Cr (5.34 w/o) alloy specimens and observed negligible length changes, associated with incipient gross surface roughening. Their data are as follows:

Thermal Cycle - 50°C to 600°C; holding times, 2 minutes cold and 5 minutes hot; transfer time between temperature zones, 5 seconds.

Number of Cycles - 10 + 90, for a total of 100.

Dimensional Changes - Negligible at 10 cycles; at 100 cycles, 0.10% to 0.65% increase in diameter, - 0.03% to 0.09% increase in length.

REMARKS

Ten cycles between 50°C and 600°C had little effect on either the dimensions or surface condition. After 100 cycles, however, roughening of the surface was clearly evident on all three specimens. The slight dimensional changes noted in some cases after 100 cycles may be attributed largely to this observed surface roughening.

Examination of the roughened surface under a low power (15x) microscope revealed generally dispersed surface cracks on both the diameter and ends of all specimens. This cracking may be the direct result of cycling, although some surface defects in the form of cracks could be noted before cycling. In any event, the cracks were more easily detected on the roughened surface of the cycled specimens.

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Figure 1. Typical Bare Specimen before Irradiation

a. Side view

Neg. No. 16,079

2X

b. End view

Neg. No. 16,080

2X

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Figure 2. Bare Specimen with Centering Spacers in Place

a. Side view

Neg. No. 16,082

2X

b. End view

Neg. No. 16,081

2X

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Figure 3. Specimen AD-1, Bare, After a Total Atom Burnup
of 0.33% (2800 MWD/T)

a. Side view

Neg. No. 16,087

2X

b. End view

Neg. No. 16,088

2X

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Figure 4. Specimen AD-2, Bare, After a Total Atom Burnup of 0.65% (5500 MWD/T). A crack can be noted at the left end of the specimen.

a. Side view

Neg. No. 16,084

2X

b. End view

Neg. No. 16,086

2X

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Figure 5. Specimen AD-6, Zirconium-Clad, After a Total Atom Burnup of 0.12% (1000 MWD/T).

a. Side view

Neg. No. 16,106

2X

b. End view

Neg. No. 16,105

2X

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Figure 6. Specimen AD-11, Stainless Steel-Clad, After
a Total Atom Burnup of 0.23% (2000 MWD/T)

a. Side view

Neg. No. 16,103

2X

b. End view

Neg. No. 16, 104

2X

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Figure 7. Specimen AD-16, Globe Iron-Clad, After a Total Atom Burnup of 0.25% (2100 MWD/T)

a. Side view

Neg. No. 16,107

2X

b. End view

Neg. No. 16,109

2X

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