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POST-IRRADIATION EXAMINATION OF
APPR FUEL ELEMENT IRRADIATION PROGRAM SPECIMENS

I. Introduction

The following pages present the results of the post-irradiation examination of specimens comprising the first three phases of the APPR Fuel Element Irradiation Test Program. This project was initiated by Oak Ridge National Laboratory in co-operation with the Army Reactors Program to determine the effects of a number of fabrication variables upon the irradiation stability of stainless steel - uranium dioxide dispersion type fuel elements.

II. Experimental Procedure

A. Materials

Specimens were fabricated to investigate the effects of the following variables upon the irradiation stability of UO_2 -SS dispersion type fuel elements:

1. The particle size of the UO_2 and stainless steel powders used in preparing the initial core compact.
2. The use of elemental or prealloyed stainless steel powders in the initial core compact.
3. The fuel concentration.
4. The method of preparation of the UO_2 powders.
5. The degree of cold reduction in the final rolling operation.

Dimensions and physical composition of the individual specimens are shown in tables of results. Miniature specimens were used in phases I and III of the program. Phase II samples were obtained from plates of a full size APPR-type element irradiated in the MTR. A more detailed description of the fabrication and composition of the individual specimens can be found in the annotated references.^{1, 2, 3}

B. Irradiations

Unfortunately a lack of suitable elevated temperature irradiation facilities prevented irradiation of specimens from this program at temperatures and pressures simulating the proposed reactor environment. However, it was felt that the need for at least preliminary irradiation damage information on APPR-type fuel elements was sufficient to justify low temperature irradiation on the specimens. As a result, all specimens were irradiated in the cooling water of the Materials Testing Reactor. The miniature specimens comprising phase I and phase III of the program were irradiated in "leaky-rabbit" type specimen holders placed in the beryllium reflector pieces. The full size element (phase II) replaced a standard fuel element in the active lattice of the MTR, and operated there for three reactor cycles (9 weeks total).

C. Post-Irradiation Examination

After irradiation, all specimens were examined for evidence of gross defects and dimensional instability. The specimens were then sectioned and samples submitted for metallographic examination and burnup analysis. Burnup determinations were obtained from Ce^{137} analysis. In some instances,

additional sections were subjected to post-irradiation annealing and then submitted for metallographic examination.

As will be shown in the discussion of the results of the investigation, much emphasis is placed on the results of the hardness measurements obtained on the core of the irradiated specimens. Although the value and interpretation of such hardness measurements have been seriously questioned in the past, some previous work on similar type fuel materials has shown what appears to be a relationship between post-irradiation core hardness and observation of loss of structural integrity by blistering, core-clad separation, and/or core fragmentation (Table I and Fig. 1). Admittedly these data were obtained on an extremely limited number of specimens; however a trend does appear to be evident. It is important to note that these data were obtained on specimens irradiated at temperatures from 380°F to 770°F, and probably does not apply to irradiations at higher temperatures where annealing of irradiation induced hardening can occur. Also, the post-irradiation core hardness range in which such failures begin to occur probably depends on such variables as UO_2 particle size, fuel loading, and matrix composition.

TABLE I

Comparison Between Core Hardness and Percentage of Core Failures Found in the Examination of
Irradiated S.S. - 30 w/o UO_2 Dispersion Fuel Element Specimens

Post-Irradiation Core Hardness - D.P.H.	351-400	401-450	451-500	501-550	551-600	601-650	651-700
Total No. of Specimens Examined	1	1	5	6	8	1	1
No. of Specimens Showing Core Fractures	0	0	0	0	4	1	1
% of Specimens Showing Core Fractures	0	0	0	0	50	100	100

Specimen Composition - 30 w/o UO_2 ($< 3\mu$ particle size)
70 w/o S.S. (18% Cr - 9% Ni - 73% Fe)

Irradiation History - Specimens irradiated in MTR in pressurized water-filled bombs at temperatures from 380°F to 770°F. Reported burnups of 6% to 40% of U_{235} .

Comparison Between Core Hardness and Percentage of Core Failures Found in the Examination of Irradiated S.S. - 30 w/o UO_2 Dispersion Fuel Element Specimens

Fig. 1

III. Results

A. Dimensional Stability

Unfortunately, accurate dimensional measurements were not obtained on these specimens before irradiation, and the as-fabricated dimensional tolerances (i.e. - Thickness - $0.030'' \pm 0.001''$) were such that thickness increases of as much as 7% could possibly be undetected. Growths of this magnitude seem unlikely however, since post-irradiation dimensional measurements failed to reveal evidence of any given dimension exceeding the as-fabricated tolerance. This was true even of specimens from phase III where burnups of over 50% of U_{235} (2.0% total core atoms) were reported.

B. Metallographic and Hardness Results

Results of the examination of the initial portion of the specimens from phase I of the program have been reported by M. J. Feldman.⁴ His data are also included here so as to present all results from the irradiation program in a final complete report.

Metallographic examination of sections from all specimens in the irradiation program failed to reveal any evidence of loss of structural integrity as a direct result of in-pile irradiation. This would tend to verify the results of the dimensional measurements which indicate that no gross dimensional changes have occurred as a result of irradiation. Blistering and core fracturing was produced in some specimen sections that were subjected to post-irradiation annealing, but this will be discussed later.

Slight changes in the structure of the UO_2 particles were noted in the examination of the irradiated specimens. The UO_2 particles which are

fragmented during fabrication of the plate (Fig. 2) apparently sinter early in the irradiation life of the element. At burnups as low as 10% of U_{235} atoms, the UO_2 particles are fully sintered and appear relatively dense (Fig. 3). At higher burnups, small pores begin to appear in the UO_2 particles, the degree to which they occur apparently being proportional to the burnup (Fig. 4 and Fig. 5). Similar observations have been reported by W. K. Barney.⁵

Complete results of the hardness measurements made on the individual specimens are shown in Tables II, III, and IV; however supplementary tables have also been prepared to facilitate the discussion of the effects of the various fabrication factors upon the irradiation stability of the fuel element.

The effect of UO_2 particle size upon the hardness of specimens from phase I of the program is shown in Table V. It will be noted that both the post-irradiation core hardness and the increase in core hardness as a result of irradiation are greater for decreasing UO_2 particle size. This is in agreement with calculations that indicate that irradiation damage to dispersion type fuel elements can be reduced by increasing the particle size of the fissile material. Evidence that this hardness increase also reflects a loss of ductility of the core matrix is shown in Table VI. Unfortunately, Feldman did not obtain burnup analysis on the even numbered specimens #12 through #26, therefore the reported burnups for these specimens were calculated.

Data on the effect of the particle size of the stainless steel powder used in the preparation of the initial core compact upon the irradiation

stability of the resultant fuel element is somewhat limited. Comparison of the post-irradiation core hardness of specimens #1 and #20, #2 and #22, #3 and #4, and #7 and #8 from phase I of the program indicates slightly higher hardnesses for specimens fabricated with the larger particle size S.S. powders. The differences are slight however, and probably can be attributed to a slight variation in the final UO_2 particle size resulting from the use of the different particle size stainless steel powders.

Data on the effect of the use of elemental or prealloyed stainless steel powders in the preparation of the initial core compact upon the irradiation stability of the resultant fuel element is even more limited. The one specimen (#9) from phase I of the program fabricated with elemental S.S. powders exhibited a significantly lower post-irradiation core hardness than a comparable specimen (#17) fabricated with prealloyed powders. The pre-irradiation core hardness of specimen #9 was also significantly lower however, and the significance of this data is therefore unknown.

The effect of varying fuel concentration upon the irradiation induced hardening of SS- UO_2 dispersion fuel materials is shown in Table II and Table III. Although specimens with increasing fuel concentration show slightly higher post-irradiation core hardnesses for approximately equivalent percentage uranium burnups; on the basis of percentage total core atom burnup, little difference in core hardness is apparent. Unfortunately a direct comparison of core hardness values does not permit a relative evaluation of irradiation damage between specimens fabricated with varying fuel concentrations. The reported core hardness values are inherently affected by the relative volume of UO_2 and stainless steel in the core, and therefore are a function of the fuel concentration itself.

It is felt that the core hardness changes noted throughout this investigation are largely indicative of hardness changes of the stainless steel matrix. Since the volume of stainless steel in the core of the specimens with the higher fuel concentration is less than that for specimens with lower fuel concentrations, a greater increase in the hardness of the stainless steel is required for the higher fuel concentration specimen to produce an apparently equivalent post-irradiation core hardness. It therefore appears that damage is more severe in elements with higher fuel concentrations, as would be expected.

The variation in post-irradiation core hardness for specimens fabricated with UO_2 powders prepared by four different manufacturing methods is shown in Table IV. It will be noted that specimens fabricated with the "C" and "D" oxide types show a significantly lower post-irradiation core hardness than specimens fabricated with the "A" and "B" oxide types. However, metallographic examination of the specimens from this group revealed that the UO_2 particle size of the "A" and "B" specimens was significantly smaller than that of the "C" and "D" specimens (Fig. 6 and Fig. 7). As a result, it appears that the method of preparation of the UO_2 powder determines its resistance to fragmentation during the fabrication process, and that the variation in post-irradiation core hardness in specimens from this phase of the program is largely a result of the subsequent variation in UO_2 particle size of the fabricated plate. Some variation in hardness might also be expected if the different oxide types exhibit a significant variation in the ability to retain fission products. Unfortunately, it is not possible to

determine from the hardness data whether this difference exists because of the variation in particle size.

Specimens from phase III of the program suprisingly showed little variation in post-irradiation core hardness as a result of varying degrees of cold reduction during the final rolling operations. One would expect significant fragmentation of the UO_2 particles during cold reduction, resulting in a decrease in the UO_2 particle size of the finished plate. Metallographic examination of the plates confirmed the fact that fragmentation of the UO_2 particles was more severe with increasing cold reduction during fabrication. This is shown in Fig. 7 (a 0% cold reduction sample) and Fig. 8 (a 30% cold reduction specimen). However, the individual fragments from each UO_2 particle remain in relatively close contact, and, upon irradiation, sinter to form UO_2 particles of approximately the same size as that existing before cold reduction. Therefore, it appears that cold reduction during the final rolling operations does not significantly effect the irradiation stability of the fuel plate, at least for the systems investigated in this program.

Although no evidence of core structural failures were noted in the examination of the as-irradiated specimens, core-cracking and blistering could be produced by subjecting sections of some of the specimens to post-irradiation annealing. For example, sections of the phase I specimens which had burnups of approximately 20% showed no evidence of loss of structural integrity when annealed at 400°C for 24 hours (Fig. 9). Sections from the same specimens annealed at 600°C for the same period of time showed

significant fracturing of both the UO_2 particles and the S.S. matrix material (Fig. 10). The extent of fracturing appeared to be slightly greater for specimens fabricated with the larger UO_2 particle sizes, but this may be a result of the greater tendency towards stringering in the larger UO_2 particle size specimens. Annealing of sections of the same specimens at 800°C resulted in very severe core fracturing in all specimens. Feldman has reported no evidence of such fracturing as a result of annealing the same type specimens irradiated to approximately 13% burnup, even at annealing temperatures of 1000°C . He does mention a large degree of "pull-out" of the UO_2 particles in metallographic preparation of the 1000°C anneal specimens. Evidently this is a result of the fragmentation of the UO_2 particles from fission gas pressure during the 1000°C anneal.

Sections of specimens from phase III of the program irradiated to over 40% B.U. of U_{235} also showed evidence of core cracking when subjected to post-irradiation annealing at 315°C (600°F). Cracking was most severe in specimens fabricated with the "A" and "B" oxide types, in most instances evidence of the initiation of blistering being noted (Fig. 11). Specimens fabricated with the "C" and "D" oxide types were significantly superior in this respect; showing rather severe cracking of the UO_2 particles, but only slight fracturing of the stainless steel matrix (Fig. 12). Specimens subjected to approximately 10% B.U. showed no evidence of cracking of either the UO_2 or stainless steel as a result of this post-irradiation anneal (Fig. 13).

Hardness variations were also noted as an effect of post-irradiation annealing of sections from specimens from phase I of the program. As shown in Table VII, an increase in the core hardness is apparent when the specimens were subjected to a post-irradiation anneal for 24 hours at 400°C , the increase being proportionally greater with decreasing UO_2 particle size. After post-irradiation annealing at 600°C , the core hardness has decreased to approximately the original post-irradiation core hardness. At 800°C , a further reduction in hardness is noted; and, at 1000°C , the core hardness is approximately that of the pre-irradiation values.

It will be noted that all specimens irradiated in this investigation contained small amounts of a burnable poison in the form of B_4C . In view of some recent data which shows rather severe irradiation damage to boron containing stainless steels, the question comes to mind of what effect such poison additions has upon the irradiation stability of the fuel element. Although the amount of boron contained in these specimens appears to be small, at 20% B.U. of uranium the burnup of boron atoms accounts for from 10-15% of the burnup of total core atoms (depending on the fuel loading). From the standpoint of damage resulting from fission gases, the effect of burnup of the boron becomes even more significant. For example, at 20% B.U. of uranium in a typical phase III specimen, over 30% of the total gases produced during irradiation is helium from burnup of the boron. Due to burn-out and decay of the true fission gases, the actual percentage of helium present at any given time is significantly greater than this.

It is known that the B_4C reacts with the stainless steel during fabrication of the fuel element. Should this reaction result in preferential segregation of the boron at the grain boundaries of the stainless steel, it would appear likely that a rather severe reduction in the mechanical properties of the matrix of the fuel element could occur. Metallographic examination of specimens subjected to post-irradiation annealing indicates that fracturing of the core originates at, or within, the UO_2 particles. Therefore it appears that this fracturing is a result of fission gas pressure within the UO_2 particles. With sufficient burnup, these pressures are sufficient to cause cracking of the stainless steel matrix. However, the pressures required to produce such cracking may be significantly lower in boron containing elements due to damage to the stainless steel matrix from burnup of the boron. Unfortunately no comparable non-poisoned fuel specimens were incorporated in the test program. As a result, it is not known what part of the observed irradiation effects can be attributed to the burnable poison.

IV. Summary

APPR-type dispersion fuel element specimens containing 17.9, 22.2, and 25.8 w/o UO_2 have been irradiated at low temperature to respective burnups of approximately 50, 20, and 20% of uranium with no evidence of gross dimensional changes or loss of structural integrity. Blistering and/or core cracking has occurred when sections of 17.9 w/o UO_2 specimens irradiated to burnups over 40% of uranium were subjected to post-irradiation annealing at 600°F for 24 hours.

Post-irradiation core hardness measurements indicate that significant differences in irradiation damage exist between the various specimen types. These data indicate that the effects of the fabrication variables investigated in this program are as follows:

1. The severity of irradiation damage in dispersion type fuel elements is inversely proportional to the UO_2 particle size of the fabricated plate. The particle size of the UO_2 powder used in preparation of the initial core compact and the method of preparation of the UO_2 powders largely determine the final UO_2 particle size of roll-bonded, dispersion fuel plates. The particle size of the stainless steel powder used in the initial core mixture and the degree of cold reduction during final sizing of the fuel plate are apparently of relatively minor importance, at least for the systems investigated in this program.

2. The severity of irradiation damage is directly proportional to the fuel concentration. Where an increased fuel loading is accompanied by an increase in the loading of the B_4C burnable poison to facilitate reactor control, the possibility of serious irradiation effects is increased to an even greater degree.

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TABLE II

Irradiation Induced Hardening of Phase I Fuel Element Specimens

Specimen No.	Plate Type*	UO ₂ in Core		S.S. Core		Burnup		Pre-Irradiation Hardness - D.P.H.		Post-Irradiation Hardness - D.P.H.	
		Type	Part. Size (μ)	Type	Part. Size (μ)	% U ₂₃₅	% Total Core Atoms	Core Ave.	Clad Ave.	Core Ave.	Clad Ave.
1	2	Sintered	44-74	Prealloyed	< 74	14.8	0.87	182	138	433	220
2	2	"	74-105	"	< 74	13.6	0.80	182	136	395	212
3	2	Steam-Treat	< 105	"	105-109	16.4	0.96	173	140	396	196
4	2	"	< 105	"	< 44	17.4	1.02	182	125	438	189
5	1	Sintered	44-74	"	< 74	22.8	0.9	186	126	424	180
6	1	"	74-105	"	< 74	20.1	0.9	186	133	394	181
7	1	Steam-Treat	< 105	"	105-109	21.2	0.84	178	126	416	215
8	1	"	< 105	"	< 44	20.1	0.79	182	138	402	217
9	2	Sintered	31-44	Elemental	< 44	17.2	1.00	153	122	398	192
10	2	"	< 105	Prealloyed	< 44	17.6	1.03	189	140	438	196
11	2	"	7-11	"	< 44	21.1	1.23	204	144	486	217
12	2	"	7-11	"	< 44	13.6 C.	0.79 C.	204	144	475	206
13	2	"	16-22	"	< 44	17.1	1.00	198	138	483	218
14	2	"	16-22	"	< 44	13.6 C.	0.79 C.	198	138	465	209
15	2	"	22-31	"	< 44	22.0	1.29	189	136	475	225
16	2	"	22-31	"	< 44	12.4 C.	0.72 C.	189	136	437	222
17	2	"	31-44	"	< 44	21.4	1.25	184	134	459	219
18	2	"	31-44	"	< 44	12.4 C.	0.72 C.	184	134	420	223
19	2	"	53-62	"	< 44	20.2	1.18	179	129	423	216
20	2	"	53-62	"	< 44	12.4 C.	0.72 C.	179	129	393	214
21	2	"	74-88	"	< 44	17.0	1.00	184	136	412	220
22	2	"	74-88	"	< 44	12.4 C.	0.72 C.	184	136	390	210
23	2	"	85-105	"	< 44	21.4	1.25	178	144	394	228
24	2	"	85-105	"	< 44	12.8 C.	0.75 C.	178	144	371	234
25	2	"	< 105	"	< 44	20.5	1.20	189	142	461	231
26	2	"	< 105	"	< 44	12.8 C.	0.75 C.	189	142	401	215

* Type "1" Plates - 17.93 w/o UO₂, 0.19 w/o B₄C, 81.88 w/o Stainless Steel Core LoadingType "2" Plates - 25.81 w/o UO₂, 0.35 w/o B₄C, 73.84 w/o Stainless Steel Core Loading

TABLE III

Irradiation Induced Hardening of Phase II Fuel Element Specimens

Plate Type		Burnup		Clad Hardness - D.P.H.		Core Hardness - D.P.H.	
		% U ₂₃₅	% Total Core Atoms*	Pre-Irrad.	Post-Irrad.	Pre-Irrad.	Post-Irrad.
15MWY	(H.F.	15.6	0.65	119	195	172	369
	(L.F.	5.4	0.22	119	166	172	307
22.5MWY	(H.F.	18.2	0.89	138	197	168	398
	(L.F.	6.5	0.32	138	178	168	340
30MWY	(H.F.	12.4	0.76	137	184	169	370
	(L.F.	5.9	0.36	137	190	169	313

15MWY Plate - 18.75 wt % UO₂, 0.21 wt % B₄C, 81.04 wt % 302 B Stainless Steel

22.5MWY Plate - 22.18 wt % UO₂, 0.27 wt % B₄C, 77.55 wt % 302 B Stainless Steel

30MWY Plate - 26.89 wt % UO₂, 0.39 wt % B₄C, 72.72 wt % 302 B Stainless Steel

* Neglecting burnup of B¹⁰ from B₄C burnable poison.

TABLE IV

Post-Irradiation Hardness of Phase III Specimens

Spec. No.	% U	Burnup % Core Atoms	Clad Hardness - D.P.H.		Core Hardness - D.P.H.	
			Pre-Irr.	Post-Irr.	Pre-Irr.	Post-Irr.
A-0	52.4	2.06	127	208	212	456
A-10	40.0	1.57	127	146	186	397
A-30	54.5	2.14	151	209	181	435
B-0	10.5	0.41	128	153	211	326
B-10	40.0	1.57	125	183	194	417
B-30	53.5	2.10	130	207	183	416
C-0	51.2	2.01	120	204	204	412
C-10	13.1	0.52	135	129	186	270
C-30	57.1	2.25	154	189	187	381
D-0	53.1	2.09	132	200	206	364
D-10	49.3	1.94	134	146	200	355
D-30	29.2	1.15	128	201	172	355

Core Composition: 17.92 w/o UO_2 (44-88 μ particle size)

0.19 w/o B_4C (minus 325 mesh)

81.89 w/o high silicon type prealloyed stainless steel (minus 100 mesh)

TABLE IV
(Continued)

UO₂ Preparation Methods

- A. Type A Specimens (Reduction of Burned Chips)
 - 1. Chips burned to U₃O₈
 - 2. Reduced to UO₂ at 1100°C for two hours
 - 3. Sintered at 1800°C for one hour
- B. Type B Specimens (Hydrogen Peroxide Precipitated)
 - 1. Peroxide precipitation of nitrate solution
 - 2. Reduced to UO₂ at 900°C for two hours
 - 3. Sintered at 1800°C for one hour
- C. Type C Specimens (Steam Oxidation of Metal)
 - 1. Steam oxidation of metal chips at 350°C
 - 2. Heated to 900°C in hydrogen
 - 3. Sintered at 1800°C for one hour
- D. Type D Specimens (Reduction of Uranium Trioxide)
 - 1. Peroxide precipitation of nitrate solution
 - 2. Converted to UO₃H₂O in autoclave at 250°C for nine hours
 - 3. Converted to UO₂ at 950°C for 30 minutes
 - 4. Sintered at 1700°C for two hours

Rolling History

- A. Type "0" Specimens - Hot rolled, no cold reduction
- B. Type "10" Specimens - 10% cold reduction
- C. Type "30" Specimens - 30% cold reduction

TABLE V

Effect of UO_2 Particle Size Upon the Irradiation Induced
Hardening of UO_2 - SS Dispersion Type Fuel Elements

UO ₂ Particle Size (μ)	Pre-Irrad. Core Hardness D.P.H.	Post-Irrad. Core Hardness - D.P.H.	
		$\approx 13\%$ B.U.	$\approx 20\%$ B.U.
7-11	204	475	486
16-22	198	465	483
22-31	189	437	475
31-44	184	420	459
53-62	179	393	423
74-88	184	390	412
88-105	178	371	394
304 S.S. Cladding	139	217	221

TABLE VI
Phase I Bend Test Results

Spec. No.	UO ₂ Particle Size	3/4" Bend Radius	1-1/2" Bend Radius
12	7-11 μ	Core Cracking	Core Cracking
14	16-22 μ	Core Cracking	No Cracking
16	22-31 μ	Core Cracking	No Cracking
18	31-44 μ	Slight Cracking	No Cracking
20	53-68 μ	-----	No Cracking
22	74-88 μ	No Cracking	No Cracking
24	88-105 μ	No Cracking	No Cracking

Note: Similar unirradiated specimens could be subjected to a bend of 105° about a 1/8" radius without evidence of core fracturing.

TABLE VII

Effect of Post-Irradiation Annealing on the Hardness of Phase I Specimens

UO ₂ Particle Size	Pre-Irrad. Core Hardness	Post-Irrad.		200°C Anneal		400°C Anneal		600°C Anneal		800°C Anneal		1000°C Anneal	
		13% BU	20% BU	13% BU	20% BU	13% BU	20% BU	13% BU	20% BU	13% BU	20% BU	13% BU	20% BU
7-11	204	475	486	--	508	586	572	500	502	364	Samples too badly fractured for reproducible hardness measurements	115	--
16-22	198	465	483	--	480	579	575	512	508	349		97	--
22-31	189	437	475	--	471	509	550	446	478	338		141	--
31-44	184	420	459	--	456	462	484	425	453	325		168	--
53-62	179	393	423	--	433	397	481	393	432	392		135	--
74-88	184	390	412	--	410	406	452	333	379	276		173	--
88-105	178	371	394	--	405	389	438	346	356	278		144	--
Typ. S.S. Cladding	139	217	221	--	221	219	224	156	155	144		148	--

Note: All specimens annealed for 24 hours at temperature.

RMG 1923

250X

Fig. 2

UO₂-S.S. Fuel Element - As Fabricated

RMG 1924

250X

Fig. 3

UO₂-S.S. Fuel Element - 10% B.U. of U₂₃₅

677 023

RMG 1940

Fig. 4

250X

UO₂-S.S. Fuel Element - 30% B.U. of U₂₃₅

RMG 1936

Fig. 5

250X

UO₂-S.S. Fuel Element - 50% B.U. of U₂₃₅

677 024

RMG 1917

250X

Fig. 6

Specimen "A-O" - As Fabricated

RMG 1935

250X

Fig. 7

Specimen "D-O" - As Fabricated

677 025

RMG 1939

250X

Fig. 8

Specimen "D-30" - As Fabricated

RMG 2445

250X

Fig. 9

Specimen No. 21
Post-Irradiation Annealed at 400°C for 24 Hours

RMG 2446

250X

Fig. 10

Specimen No. 21
Post-Irradiation Annealed at 600°C for 24 Hours

RMG 2447

250X

Fig. 11

Specimen "B-30"
Post-Irradiation Annealed at 600°F for 24 Hours

RMG 2448

250X

Fig. 12

Specimen "D-30"
Post-Irradiation Annealed at 600°F for 24 Hours

RMG 2449

250X

Fig. 13

Specimen "C-30"
Post-Irradiation Annealed at 600°F for 24 Hours

677 028

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