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ANNUAL PROGRESS REPORT ON FUEL ELEMENT DEVELOPMENT

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SUMMARY

The Reactor Fuels and Materials Development Program is a project aimed at the improvement of fuel elements for high flux test reactors. The investigation of new fuel compositions, distributions, and geometries is being undertaken in order to increase fuel life, to improve the flux distribution and to provide a means of safely reaching higher reactor operating power and power density in these reactors. A study of the effects of nuclear irradiation on the fuel and structural materials is being made in order to predict the performance of these materials in more advanced reactor designs.

The present report summarizes the past year's progress to June 30, 1962, and describes the fabrication and irradiation of samples containing up to 50 weight percent uranium-aluminum alloys, cermets of UO_2 , U_3O_8 , UC, UN, U_3Si and aluminum, clad in various aluminum and beryllium-aluminum materials. The use of ThO_2 and Th cores, the addition of BeO to cermet cores and high density fuel cores of U-Al intermetallics produced by powder metallurgy techniques have been studied during the year. High strength APM claddings involving Al_2O_3 contents from 6 to 10% have been tested and indicate the need for improved quality control of the APM material. Duplex claddings involving burnable poison layers and APM clad with corrosion resistant X8001 have shown promise where special properties are desired.

The results of the work continue to demonstrate the excellent radiation stability of uranium-aluminum fuels even after long irradiation exposure at elevated temperatures. Tests up to 350°F and after 50% burnup of the U235 in uranium-aluminum alloys, show no appreciable dimensional or microstructure changes. UO_2 and U_3O_8 react with aluminum under radiation to form UAl_4 .

Tensile tests of these fuels at ambient temperatures show appreciable loss in ductility with irradiation; several compositions actually exhibiting zero ductility. Irradiation at temperatures up to 200°F of cold-worked and of heat-treated aluminum does not destroy the pre-irradiation hardness and strength of these materials.

Computer optimization of fuel element geometries from the standpoint of heat transfer, hydraulics, and strength has resulted in the design and fabrication of a 32-plate fuel element. Hydraulic tests produced favorable results and the element is ready for MTR testing.

Future work stressing materials development will attempt to extend uranium-aluminum fuels to higher temperatures (400-800°F), will continue the studies on graded fuels, beryllium damage, and the thorium-U233 system. Tensile testing will be extended to higher temperatures in pre- and post irradiation measurements and the study of the effect on cold worked and tempered materials of elevated temperature-radiation exposures will be continued.

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I. INTRODUCTION

This report summarizes the progress in Reactor Fuels and Materials Development at the MTR/ETR for Fy 62. The previous report⁽¹⁾ in this series covered the work performed primarily in Fy 61 but included some data to August 1961. The early work⁽¹⁾ ⁽²⁾ has included the successful fabrication and irradiation of fuel plate samples containing up to 45% uranium alloyed with aluminum. In the past year, this work has been extended to 50 wt% uranium-aluminum alloys, and has included cermets of UC, UN, U₃Si, and ThO₂ in aluminum. A technique for preparing high weight percent uranium-aluminum alloy powder dispersed in aluminum has been developed. The testing of prototype fuel elements is also described.

In format, the report follows what appears to be a logical division of materials studies, both fabrication and irradiation, and geometry optimization. An associated project on In-Pile Fatigue added during the last quarter of the fiscal year under the same budgeted activity will be reported separately.

II. MATERIALS

Table I is a list of the new fuel plate compositions made since the last report was written. These compositions were selected to further reactor technology by studying the effects of additional types of materials for fuel plate cores and claddings. Beryllium, for example, is being studied because of its superior moderator properties when compared with aluminum. Its low absorption cross section for neutrons would be advantageous in an MTR type reactor. The aluminum powder metallurgy products (APM's) are attractive as cladding materials due to their excellent mechanical properties at elevated temperatures. The use of uranium-aluminum intermetallic compounds dispersed in aluminum alloys by means of powder metallurgy techniques, is being studied because of the potential stability of these materials in the reactor environment. Other fuel plate compositions under study can be found in a prior report.⁽¹⁾

In addition to studying the feasibility of fabricating these compositions, studies have been made to determine how well they withstand service in the reactor. For example, a study on the effects of irradiation and temperature on the tensile properties of fuel plates has been started. Further investigations have been made on corrosion resistance of aluminum materials in the reactor and on the stability of uranium oxides.

To assist in determining the fuel plate core geometry required for graded fuel elements, gamma scans of the fuel plates from two MTR and two ETR fuel elements were made.

Finally, a number of sample fuel plates which failed while undergoing irradiation testing have been examined and the cause of failure determined.

A. Fabrication

1. Core Materials

The materials used in the cores of the sample fuel plates are listed in Table I. The preparation of each of these materials is discussed below.

a. U-Al Alloys

These alloys were all prepared by vacuum induction melting of fully-enriched uranium metal, aluminum ingot, reuseable scrap materials and the required alloying additions. The alloys were cast into copper molds. After soaking at 1100°F for at least one hour, the ingots were hot rolled to the thickness required for blanking. This thickness, of course, was dependent upon the final core thickness specified for the finished plates.

The core alloys which contained 6061 and X8001 aluminum were made to determine the possibility of remelting reject fuel plates which were clad with these materials. Typical fuel plate specifications do not permit small amounts of elements such as Fe, Ni, Mg, Cu or Cr in the core alloys. These elements would be introduced by the remelting of

TABLE I

SAMPLE FUEL PLATE COMPOSITIONS

Item Number	CORE				CLADDING	
	Fuel Material (1)	Wt% Fuel	Additions	Thickness (Mils)	Material	Thickness (Mils)
78	U in Al Alloy ↓	18		20	APM M583 Al	15
76		18		20	APM M470 Al (2)	15
80		18		20	2% B as B ₄ C (Binal)	1
					1100 Al	14
81		18	6061 Al	20	6061 Al	15
82		18	X8001 Al	20	X8001 Al	15
89		18		20	50% B-Al Alloy	15
88		30		10	Beryllium	20
74		50	3 w/o Sn	8	APM M257 Al	12
75		50	3 w/o Sn	8	APM M257 Al (2)	12
					X8001 Al	5
77		50	3 w/o Sn	8	APM M470 Al	12
79		50	3 w/o Sn	8	APM M583 Al	12
87	UO ₂ in Al	32	8 w/o BeO	20	1100 Al	15
90	Thorium	100	1.5 w/o U	70	APM M257 Al	15
91	ThO ₂ in Al	63.5	1.5 w/o UO ₂	70	APM M257 Al	15
92	UC in Al	47		8	APM M257 Al	20
93	UN in Al	47		8	APM M257 Al	20
94	U ₃ Si in Al	47		8	APM M257 Al	20
114	UAl ₃ in Al	35	6061 Al	20	6061 Al	15
113	UAl ₃ in Al	35	X8001 Al	20	X8001 Al	15

Notes:

- (1) All uranium is fully enriched
 (2) Duplex cladding

plates clad with 6061 or X8001 Al alloys. No particular problems were encountered in melting, casting, or rolling the core alloys. However, the materials were more difficult to blank and to roll into fuel plates after having assembled the cores with picture frames and cover plates.

The sample fuel plates containing cores with high weight per cents of uranium alloyed with aluminum which were made in the earlier phases of this program used silicon to suppress the formation of UAl_4 . Since silicon causes trouble in the chemical reprocessing of fuel elements, tin was substituted in order to suppress the peritectic reaction in the high weight per cent uranium-aluminum alloys. These alloys were found to have the same fabrication characteristics as those containing silicon.

b. UO_2 plus BeO in Al Powder(3)

This composition, as well as the compositions containing uranium and thorium cermet cores which are described in the following paragraphs, were fabricated under a sub-contract with the Sylcor Division of the Sylvania Electric Company (later a part of General Telephone and Electronics Laboratories).

The plates in which this type of core material was used were made by roll bonding. The UO_2 with a uranium content of 87.95 wt%, was the high fired grade made by Davison Chemical Division of the W. R. Grace Company. It had a particle size of minus 140 plus 325 mesh. The BeO powder used minus 200 mesh, and the aluminum powder used was Alcoa atomized aluminum powder grade 101, - 200 mesh.

c. UC in Al Powder(3)

The UC used for these cores was made by arc melting uranium metal and graphite. Metallographic examination indicated that the UC was essentially single phase UC. The chemical analysis indicated a uranium content of 95.3 wt%.

The material was ground to minus 200 mesh, blended with grade 101 aluminum powder and compacted at 45 tsi in a $3/4"$ x $1\ 3/4"$ die. The green compacts were vacuum out-gassed at 600° C. These were then cleaned by wire brushing and assembled with a picture frame and cover plates for pressure bonding. After pressure bonding, the plates were hot rolled to finished size.

d. UN in Al Powder(3)

The UN for these cores was made by nitriding of uranium metal in ammonia. X-ray diffraction indicated that the material was mainly single phase UN.

The UN was ground to minus 200 mesh, blended with grade 101 aluminum and compacted at 45 tsi. The green compacts were vacuum out-gassed, pressure bonded to cover plates and picture frames, then hot rolled to finished size.

e. U₃Si in Al Powder⁽³⁾

The U₃Si used in these cores was made by arc melting uranium and silicon metal. The arc melted material was homogenized at 850°C for one week and then ground. Grinding was difficult. Ball milling was tried at first, but did not produce sufficiently fine material. Consequently, the material was ground in a mortar and pestle to minus 100 mesh powder but resisted further grinding to a finer particle size. Compacts of this material and 1100 aluminum powder were prepared at 45 tsi and fabricated into fuel plates using the same procedures as used for the UN and UC cermet cores. Metallographic examination of these plates disclosed nonuniform fuel distribution with agglomeration of fuel particles and uneven core thickness which exceed specifications for normal MTR/ETR fuel plates.

f. ThO₂ + UO₂ in Al Powder⁽³⁾

The same fabrication procedures were used for the plates containing ThO₂ + UO₂ in aluminum powder as were used for those containing UC, UN, and U₃Si. The ThO₂ used was minus 100 plus 270 mesh powder. However, about 40% of the plates produced were rejected due to voids.

g. 1.5 wt% U in Th Alloy⁽³⁾

To date, no satisfactory procedure has been developed for making aluminum clad plates of this composition. Pressure bonding techniques have been tried, but fail to produce reliable bonds. Roll bonding seems to work better, but the cores tend to break through the cover plates.

h. High Weight Per Cent U-Al Alloy Powders

The use of high weight per cent uranium-aluminum alloy powders for fuel dispersions in the cores of fuel plates is being studied because of their potential stability in the reactor environment.

It has been shown that the uranium oxides in contact with aluminum are unstable at elevated temperatures⁽⁴⁾ and are potentially unstable under irradiation.^(1,2) When using an alloy core material, it is difficult to obtain a uniform dispersion of fuel and poison. However, if a high weight per cent uranium-aluminum alloy is powdered and used as the fissile phase with an aluminum alloy powder being used as the matrix material, then the advantages of both the alloy and cermet can be obtained.

The cores of the samples made to investigate this type of fuel plate had 68.7 wt% U-Al alloy powder diluted with either 6061 or X8001 aluminum alloy powder so as to give 35 wt% uranium in the core. The matrix aluminum and the cladding material were the same (ie., 6061 or X8001) in any one sample. The uranium was fully enriched. 0.19 wt% B₄C was blended into the core mixture to serve as a burnable poison.

By means of X-ray diffraction, the uranium-aluminum alloy powder was determined to be a mixture of the intermetallic compounds UAl₃ and UAl₄. This material was used since the preliminary development work

with depleted uranium alloys had indicated that those alloys containing over 70 wt% uranium were very pyrophoric. No problems of this nature were encountered with this alloy.

In Figure 1, a photomicrograph of the core of a finished fuel plate is shown. In this micrograph the largest particle is about 0.0015 inches in diameter. One of the advantages of the use of inter-metallic compound powders instead of wrought alloys for core materials is the ease in which the particle size can be controlled.

HF Etched

500 X

Figure 1 Cross-section of Fuel Plate Core Containing Uranium-Aluminum Alloy Powder, B_4C Powder and 6061 Aluminum Alloy Powder.

2. Cladding Materials

In Table I, the materials used as claddings for the sample fuel plates fabricated during the past year are listed. The nominal chemical compositions of these materials are given in Table II.

a. Aluminum Materials

The new aluminum-based material which has been investigated is APM-M-583. This material, with an oxide content intermediate between that of APM-M-257 and APM-M-470, was also found to be easily fabricated into fuel plate claddings.

Some of the batches of APM material used for these plates

were of lower quality than had been supplied in the past. Considerable, difficulty was encountered with these lots because of blistering, inclusions, and pitting of the APM.⁽³⁾ It is expected that as these materials become commercially available, these defects will no longer be troublesome.

With the exception of the plates with thorium metal cores, all of the aluminum-clad plates were made by roll bonding. The procedures used have been described in a previous report⁽²⁾. The plates containing thorium cores were made by pressure bonding⁽³⁾ since attempts to roll bond plates similar to these plates had failed due to the formation of brittle thorium-aluminum intermetallic compounds at the core-cladding interface.

TABLE II

NOMINAL CHEMICAL COMPOSITION OF CLADDING MATERIALS^(5, 6, 7)

Material	Weight Percent Present											
	Al	Al ₂ O ₃	Be	BeO	C	Cr	Cu	Fe	Mg	Mn	Ni	Si
APM M257 Al	Bal	6										
APM M470 Al	Bal	10										
APM M583 Al	Bal	7										
1100 Al	99+											
6061 Al	Bal					0.25	0.25		1.0			0.6
X8001 Al	Bal							0.5			1.0	
Beryllium	0.04		Bal	1.5	0.1	0.01		0.1	0.02	0.01	0.02	0.03
Be-Al Alloy	Bal		50									

b. Beryllium and Beryllium-Aluminum

The beryllium used as a cladding material was in the form of QMV beryllium hot-rolled strip. As this was a rather brittle material, pressure bonding at 600°C and ⁽²⁾ tsi was used to fabricate the plates in which this material was used.⁽³⁾ In the first plates made, 1100 aluminum was used as the picture frame material. This was not successful as the bond between these two dissimilar metals was weak.

The 50/50 Be-Al alloy used as a cladding material was too brittle to be rolled at 1100°F. Therefore, the plates using this material for cladding were fabricated by pressure bonding. 1100 aluminum was used as the picture frame material. This technique was not satisfactory as about three-quarters of the plates were rejected.

In both beryllium and beryllium-aluminum claddings, the pickling required to prepare the surfaces for bonding also tended to pit the outer surface beyond acceptable limits. Completely satisfactory procedures to produce beryllium clad plates have not been developed at this stage; however, the successful accomplishment of this objective is not considered to require major additional effort.

c. Duplex Claddings

Two of the sample fuel plate compositions made had duplex claddings. In one composition, the extra layer of material was added as a means of uniformly dispersing a burnable poison. In this composition, a 1-mil thick strip of aluminum containing a dispersion of 2 wt% B_4C was used under the 14-mil thick 1100 aluminum cladding material.

In the other duplex clad plates, APM-M-257 was alclad with X8001. The object here was to produce fuel plate claddings with the high temperature corrosion resistance of the X8001 aluminum and the high temperature strength of the APM material. In both cases, the usual fabrication techniques were used to produce satisfactory sample fuel plates.

3. Fuel Element Assembly by Roll Swaging

In order to provide a means of assembling fuel sections for hydraulic and static pressure testing, roll swaging equipment was constructed.

In Figure 2 is shown a cross-sectional view of a typical joint produced between the fuel plate and the side plate on this machine. The "fuel plate" used here is a solid aluminum plate.

M-930

As Polished

9.4 X

Figure 2 Joint Between Fuel Plate and Side Plate Produced by Roll Swaging.

B. Irradiation Effects on Sample Fuel Plates

Information from metallurgical analysis of fission effects has been collected by various techniques including electron microscopy, heat treating, metallography, mechanical testing and qualitative analysis by X-ray diffraction and electron probe. It has been shown that aluminum clad uranium-aluminum alloy and uranium-oxide aluminum dispersion type fuel plates have good dimensional stability when exposed to a variety of irradiation and intermediate (e.g. 100-250°F) temperature conditions.

Although warpage in plates during processing has been attributed to the reaction between uranium oxide and aluminum, this reaction has been taken to completion by heat treating without attendant deformation. Analysis of the reacted particles confirmed the formation of UAl_4 and UAl_3 ; suspected Al_2O_3 has not been confirmed. Since similar reaction zones appear in fissioned fuel plates, the thermal reaction studies are included. All the conditions under which the reactions take place and the relation between reaction and deformation have not as yet been established.

Studies on non-fuel containing aluminum specimens, IDO-16628,⁽⁸⁾ which show very little change in mechanical properties due to radiation, suggest that the large change in mechanical properties which occur in fuel-containing aluminum plates is chiefly due to fission damage.

1. Mechanical Testing

Mechanical testing at present includes tensile tests, microhardness surveys, and bend tests.

The current tensile test program for sample fuel plates considers the effect of various cladding materials, testing temperatures, fuel contents, type of fuel, enrichment, burnup, and additions of burnable poisons. To date, room temperature testing has been completed on some of the compositions (see Table VIII in the appendix).

Microhardness measurements have been made on the sample fuel plate micro-constituents to detect individual constituent changes that occur as a result of fissioning. Current practice also includes photographing microhardness indentations in such a way that changes in hardness from the center of the plate to the outside edge could be detected. The results of microhardness tests, some of which have been previously reported, are shown in Table IX in the appendix.

Analysis of data from Tables VIII and IX shows that 40% of the irradiated plates had zero elongation in a one inch gauge length. The specimens having zero ductility were scattered through the ranges of compositions tested and levels of radiation attained. The change in per cent elongation, due to fissioning, ranged from -58% to -100%, with a maximum per cent elongation in one inch of 4.5% for any one fissioned specimen. For the specimens which had some ductility, it is found that (1) ductility decreases rapidly due to fission damage (Figures 3, 4, 5), (2) strength decreases with an increase in fuel content (Figure 6), (3) for a given

fuel content, alloys are stronger than oxides, (4) in initial stages of irradiation, the amount of strength increase depends on the type of cladding material, and (5) increase in strength is proportional to the per cent increase in hardness of the cladding material (Figure 3).

Figure 3 Ductility vs Radiation for an 1100 Aluminum Clad Fuel Plate Containing an 18.7 w/o UO_2 in Aluminum Cermet Core.

There were no detectable effects on hardness or strength resulting from the small additions of burnable poisons. In a previous report, IDO-16727, it was shown that there were no observable microstructure changes.

Figure 4 A Uranium-Aluminum Alloy (6061 aluminum clad) Fuel Plate Tensile Specimen Embrittled as a Result of Fission Damage.

Figure 5 A Uranium Oxide (1100 aluminum clad) Sample Fuel Plate Tensile Specimen Embrittled as a Result of Fission Damage.

2. Stability of Microconstituents and Dimensions

Sample fuel plates exposed to heat treatments, irradiation at MTR conditions, irradiation at elevated temperatures and post-irradiation heat treatments have been examined for dimensional stability, microstructural changes and composition of microconstituents.

As has previously been reported⁽¹⁾ on a large number of tests, plates with cores of UO_2 dispersed in aluminum and uranium-aluminum alloy had good dimensional stability when exposed to MTR irradiation conditions. The tests reported herein for those plates subjected to the various elevated temperature conditions (Table III) also indicate good dimensional stability.

a. Microhardness

It has been found that significant hardness increases occur in the cladding, matrix, and intermetallic particles of the alloy plates due to irradiation at reactor conditions. The effect appears the same for the matrix and intermetallic particles of alloy fuel plates which are irradiated at elevated temperature or post-irradiation annealed. The claddings of the alloy plates used for elevated temperature tests, however, showed a smaller increase in hardness. A similar effect is seen when considering the matrix and cladding of the U_3O_8 + Al cermet plates. Particles of UO_2 , in UO_2 + Al cermet plates, show a decrease in hardness due to irradiation (perhaps because of porosity) at reactor conditions, but show an increase in hardness in the elevated temperature tests. The cladding of plates as indicated by Tables III and IX again did not show as much hardness increase as did plates irradiated at reactor conditions. The matrix of the plates containing UO_2 irradiated at reactor conditions showed a wide variation in hardness increase. Comparisons of these with a limited number of elevated temperature tests were not made. But, it appears that irradiation increases the matrix hardness of U-Al alloy and plates containing U_3O_8 more than it does plates containing UO_2 .

b. Metallography

The U-Al alloy plates, show only slight microstructural changes when irradiated. (See Figures 8, 9, 10). Elevating the core temperature during irradiation to a maximum of 350°F had no observable effect (Figure 11). The post irradiation anneal specimen which reached 650°F showed multi-shaded particles (Figure 12).

It has been found that the U_3O_8 particles react considerably with the aluminum matrix during irradiation (Figures 13, 14, 15) but there is no appreciable change due to a post-irradiation anneal at 500°F (Figure 16). This series of tests did not include an elevated temperature irradiation of a U_3O_8 plate.

Figures 17, 18, 19, 20, 21, 22, and 23 are photomicrographs of UO_2 particles dispersed in aluminum matrices which after fabrication have been (1) heat treated to produce reaction between UO_2 and aluminum (2) irradiated to various burnups at 200°F, (3) irradiated at 425°F, (4) post-irradiation annealed at 390 °F. The plate which was irradiated at elevated temperature (Figure 22) did not have the zoned and porous particles

TABLE III

CHANGE IN HARDNESS AND THICKNESS OF FUEL PLATES EXPOSED TO VARIOUS ENVIRONMENTAL CONDITIONS

Type Fuel Plate	Plate Exposed to these Environmental Conditions	Average DPN Hardness			% Thickness Change by Environmental Conditions	Microstructure shown in Figure Numbers
		Clad	Matrix	Particle		
32% w/o fully enriched UO_2 in Al	As fabricated	26	27	320		17
18 w/o fully enriched U-Al alloy	As fabricated	33	43*	43*		8
22.3 w/o fully enriched U_3O_8 in Al	As fabricated	26	30	140		13
42 w/o fully enriched UO_2 in Al	Heat-treated 40 hrs. at 1110°F	29	32	500 dark 570 light	+ 2	18
16.8 w/o fully enriched UO_2 in Al	Irradiated to 30% burnup	39	43	200	- 1	19
24 w/o fully enriched UO_2 in Al	Irradiated to 45% burnup	30**	56	230	+ 4	20
23.8 w/o fully enriched UO_2 in Al	Irradiated to 40% burnup	38	120	240	+ 4	21
18 w/o fully enriched U-Al alloy	Irradiated to 51% burnup	55	180	210	- 1	9

*Particles too small for individual measurements - average values

**Appears to be low according to other data in Table IX

TABLE III (Con't)

Type Fuel Plate	Plate Exposed to these Environmental Conditions	Average DPN Hardness			% Thickness Change by Environmental Conditions	Microstructure shown in Figure Numbers
		Clad	Matrix	Particle		
18 w/o fully enriched U-Al in Al	Irradiated to 69% burnup	72	200	240	0	10
22.3 w/o fully enriched U ₃ O ₈ in Al	Irradiated to 25% burnup	35	150	-	0	14
22.3 w/o fully enriched U ₃ O ₈ in Al	Irradiated to 52% burnup	59	160	-	+ 4	15
24 w/o fully enriched UO ₂ in Al	Irradiated to 18% Burnup at 425°F	28	98	470	- 2	22
18 w/o fully enriched U-Al alloy	Irradiated to 18% Burnup at 350°F	28	130	240	- 2	11
22.7 w/o fully enriched UO ₂ In Al	Irradiated to 29% burnup and post-irradiation annealed at 390°F for 2 hours	34.4	90	590	+ 2***	23
18 w/o fully enriched U-Al alloy	Irradiated to 50% burnup and post-irradiation annealed at 650°F for 4 hours	30	190	240	+ 4***	12
22.3 w/o fully enriched U ₃ O ₈ in Al	Irradiated to 54% burnup and post-irradiation annealed at 500°F for 3 hours	34.4	160	390	+ 6***	16

This change in thickness occurred during irradiation; the before and after heat-treating thicknesses were the same.

Figure 6 Strength vs Radiation for Fuel Plates of Various Compositions .

Figure 7 Increase in Plate Strength vs Percent Increase in Cladding Hardness for Plates Clad with 1100 Aluminum and Having Various Core Compositions.

HF Etched 520X
 Figure 8 Microstructure of an 18 w/o U-Al alloy, as Fabricated.
 M-569 22-15

HF Etched 500X
 Figure 9 Microstructure of an 18 w/o U-Al alloy, Irradiated to 51% Burnup.

HF Etched 500X
 Figure 10 Microstructure of 18 w/o U-Al Alloy, Irradiated to 69%
 Burnup.
 M-924 19-5

As Polished 500X
 Figure 11 Microstructure of 18 w/o U-Al Alloy Irradiated to a Burnup
 of 18% at 350°F.

As Polished 500X
 Figure 12 Microstructure of 18 w/o U-Al Alloy, Irradiated to 50%
 Burnup and Post-Irradiation Annealed at 650.

M-685 47-12

As Polished 500X
 Figure 13 Microstructure of 22.3 w/o U_3O_8 in Al Cermet, As Fabricated

As Polished 500X
 Figure 14 Microstructure of 22.3 w/o U_3O_8 in Al Cermet Irradiated to 25% Burnup.

M-802 48-11

As Polished 500X
 Figure 15 Microstructure of 22.3 w/o U_3O_8 in Al Cermet, Irradiated to 52% Burnup.

As Polished 500 X
 Figure 16 Microstructure of 22.3 w/o U_3O_8 in Al Cermet, Irradiated to 54% Burnup, and Post-Irradiation Annealed at 500°F for 3 hours.

As Polished 500 X
 Figure 17 Microstructure of 32 w/o UO_2 in Al Cermet, as Fabricated.

As Polished
 Figure 18 Microstructure of 42 w/o UO_2 in Al Cermet, Heat Treated
 40 Hours at 1110° F.

520 X

M-675 52-6

As Polished
 Figure 19 Microstructure of 16.8 w/o UO_2 in Al Cermet, Irradiated to
 30% Burnup.

500 X

As Polished 500 X
 Figure 20 Microstructure of 24 w/o UO_2 in Al Cermet, Irradiated to
 45% Burnup. M-716 8-17

As Polished 500 X
 Figure 21 Microstructure of 23.8 w/o UO_2 in Al Cermet, Irradiated to
 40% Burnup.

M-908

As Polished 500 X
Figure 22 Microstructure of 24 w/o UO_2 in Al Cermet, Irradiated to 18% Burnup at 425°F.

M-927

As Polished 500 X
Figure 23 Microstructure of 22.7 w/o UO_2 in Al Cermet, Irradiated to 29% Burnup and Post-Irradiation Annealed at 390°F for 2 Hours.

typical of plates irradiated at MTR conditions (Figures 19, 20, 21). Because of the somewhat unexpected appearance of the microstructure more tests, under similar conditions, are being made in order to confirm the results. Particles thermally reacted have been identified as UAl_4 (grey) UC or UN (dark) by electron probe analysis. UAl_3 has been detected in some of the thermally-reacted specimens by X-ray diffraction.

In order to attempt to determine the damage mechanisms associated with the irradiation of a UO_2 -aluminum dispersion, a sample fuel plate (51-9) was examined by the Battelle Memorial Institute.* This plate had originally contained a core of 11.6 w/o UO_2 dispersed in aluminum. The fabricated procedures used to produce this plate have been described in IDO 16727⁽²⁾. The plate had been irradiated to a burnup of 50% of the original U-235. The results of this investigation are summarized in the following excerpts from the BMI report⁽⁹⁾.

"The irradiated fuel plate was examined and sectioned for metallographic examination. The sample was mounted flat in "Hysol" plastic and was ground to the centerline on 600 grit SiC paper. It was then mechanically polished using one micron diamond dust on Miracloth with a Syntron vibratory polisher. The sample was lightly etched with 95 cc H_2O , 1.5 cc HNO_3 , 1.5 cc HCl , and 1.0 cc Hf . An unirradiated sample was prepared in the same manner for comparison. The samples were examined and photomicrographs obtained in the as-polished and etched conditions on the light microscope. Figures 24, 25, and 26 illustrate the appearance of the material in the unirradiated condition, in the irradiated as-polished condition, and in the irradiated etched condition, respectively. The porosity in the UO_2 as shown in Figures 25 and 26 is probably due to fission gas agglomeration and indicates that the UO_2 has received a relatively high burnup. Two distinct layers between the UO_2 and aluminum are apparent both in the as-polished and as-etched structures. In the as-polished condition a dark layer lies immediately adjacent to the UO_2 with a light-appearing layer visible outside this. When etched these same layers are seen with the exception that the outermost layer is apparently etched out. Voids are visible in all three layers. The original angularity of the UO_2 particle appears to be associated with the total particle and layers while the UO_2 in the center is gradually assuming spherical form. This suggests some diffusion controlled reaction between the UO_2 and aluminum.

"Plastic replicas of the irradiated and unirradiated samples were taken in both the as-polished and etched conditions. All of these replicas were shadowed with platinum in a vacuum evaporator and carbon was deposited normally onto the shadowed surfaces. The composite replicas were cut into appropriately sized pieces, mounted on specimen screens, and the plastic dissolved away leaving a shadowed carbon replica.

"Figure 27 shows a relatively small area of Al in the lower portion of the micrograph. (Notations of the UO_2 and Al phases have been made on the prints.) The UO_2 is smoother than the Al and is protruding above the surface of the Al. The boundary between the UO_2 and the Al is free of obvious third phases. This sample was not etched.

* An electron microscope and "hot" X-ray diffraction equipment have since been installed by Phillips. Consequently, examinations of this type are now being undertaken at this site.

As Polished

(a) Flat Section

250X

As Polished

(b) Edge Section

250X

BMI-RM-20846

Etched (light)

(c) Flat Section

250X

Figure 24 Unirradiated UO_2 Dispersed in Aluminum

BMI-HC-9381

As Polished

500X

(a) Flat Section

BMI-HC-9382

As Polished

Approx. 890X

(b) Flat Section

Figure 25 Unirradiated UO_2 Dispersed in Aluminum.

Figure 26 Irradiated UO_2 Dispersed in Aluminum

As Polished

17,500 X

Figure 27 Unirradiated Control Sample of UO_2 Dispersed in Aluminum.

"Figure 28 shows the surface of the polished and etched unirradiated specimen. The UO_2 does not seem to be markedly pitted by the etchant while the Al is extensively pitted. There was no evidence of a grain boundary phase between the UO_2 and the Al.

"Figure 29 shows the polished surface of the irradiated specimen. A number of changes have occurred. The most obvious change is the appearance of relatively large circular pits in the surface of the UO_2 particles. These are probably due to the agglomeration of fission gas. In the upper portion of Figure 29, the Al surface is shown. It is somewhat roughened and shows scratch marks. It seems to be at nearly the same level as the UO_2 surface, however, this is very difficult to judge because of the great depth of focus of the electron microscope. Between the Al and the UO_2 there is a slightly raised area varying in width from 1.5 to 4 μm at this magnification in which there is a slight suggestion of small circular pits. This area is not sharply delineated but it seems to be present in all the micrographs of this specimen. Comparison with the light micrographs (Figure 25) is difficult but this area apparently corresponds to the darker layer adjacent to the UO_2 particle. The lighter area adjacent to the aluminum as seen in the light micrograph is not readily apparent in the electron micrographs. Only in Figure 29 does there appear a suggestion of an additional layer lying below the raised portion. A few large pits or voids are seen in this area. Part of the difficulty is undoubtedly due to smearing of the structure during polishing making it difficult to clearly delineate the various layers.

"Figure 30 shows the surface of the irradiated specimen after polishing and etching. The various areas are clearly shown in Figure 30 and correspond with the regions seen in the light micrographs of Figure 26. At the bottom is the polished area of the UO_2 showing fission gas bubbles or pits. Next is shown the first layer with more numerous but smaller sized gas voids or pits. Next there is a severely attacked region which is etched out in the light micrographs. Finally the surface of the Al matrix appears at the top of the micrograph.

c. X-Ray Diffraction

The metallographic samples were then examined by X-ray diffraction techniques for phase or structural changes. X-ray diffraction lines of aluminum were observed from the irradiated sample. There was also an extremely diffuse diffraction band at the first two lines of UO_2 which were too weak for measurement. No other phases could be detected. These results tend to verify that the UO_2 was burned up to a high degree. Work performed by R. M. Berman(10) indicates that UO_2 becomes almost completely amorphous at 30×10^{20} fissions per cc.

"There was no evidence of any Al_2O_3 or any other phase having been formed in this material. This could be due simply to insufficient quantities for detection although even if Al_2O_3 were present, it is very unlikely that X-ray diffraction could detect the Al_2O_3 lines. Evidence has been reported by other investigators indicating that the crystal structure of Al_2O_3 (in some cases) is destroyed at 10^{16} fission fragments per cc. Since a large portion of the fission fragments would escape from the fuel particle itself, the crystal structure of any Al_2O_3 formed around

Etched

17,500 X

Figure 28 Unirradiated Control Sample of UO_2 Dispersed in Aluminum.

As Polished

16,000 X

Figure 29 Irradiated Sample of UO_2 Dispersed in Aluminum.

Etched

16,000 X

Figure 30 Irradiated Sample of UO_2 Dispersed in Aluminum.

the UO_2 particle would probably be destroyed by fission fragment recoil damage, making it impossible to detect by X-ray diffraction techniques.

d. Conclusions

"Two layers form at the interface between UO_2 and aluminum during irradiation. The identity of these layers is unknown. However, some conjecture concerning these layers is possible. They appear to form by a diffusion controlled process between the UO_2 and aluminum with UO_2 being consumed in the process. The outer layer, adjacent to the aluminum, which was removed by etching, is more reactive than the inner layer. This suggests an aluminum-uranium compound, possibly UAl_3 ; this phase has been observed in irradiated UO_2 -Al dispersions. The inner layer, which is harder than the UO_2 since it appeared raised with respect to the UO_2 in the as-polished condition, is probably Al_2O_3 .

"On the other hand, the appearance of a rim around UO_2 has been observed in stainless steel- UO_2 dispersions which is similar in nature to the dark layer adjacent to the UO_2 in these samples.⁽¹¹⁾ It is observed when burnup on the order of 50 a/o of the uranium are achieved. This rim has never been identified but may result from segregation of fission-product oxides out of the UO_2 . A recoil zone outside this rim is also observed in stainless steel- UO_2 dispersions, although in stainless steel this zone is much more uniform in width than the outer light layer in these samples." The microstructure of the irradiated UO_2 dispersed in aluminum is very similar to that of irradiated UO_2 dispersed in stainless steel of BeO, i.e., the UO_2 particles appear to have been liquid. The irradiation temperatures for the UO_2 -Al dispersions were low, $\approx 300^\circ\text{F}$. This would tend to confirm the assumption made by Belle⁽¹¹⁾ "that the amorphous UO_2 formed upon fissioning shows liquid-like properties with respect to exhibiting a pressure-dependent solubility for the noble gases."

3. Gamma Scanning of Fuel Plates

It has long been of interest to determine just how closely the fuel burnout followed the vertical neutron flux distribution measured by flux wires in the water channels between the fuel plates. Particularly of interest is the flux peaking measured at the lower end of the fuel element and which is usually the basis for locating the hot spot in this area. IBM computations using FLIP-1 codes have indicated that large disadvantage factors exist between fuel core and water channel. Somewhat inconclusive results have been obtained by scanning entire fuel elements in the MTR canal. This work would be completely suitable for determining the geometry required for the various loadings of a graded fuel plate.

The project consisted of milling the side plates from irradiated fuel elements and then gamma scanning the individual fuel plates. This was accomplished by passing the plate at a controlled speed in front of a $1/8$ " opening in a plug through a hot cell wall. The relative gamma intensities through the opening were then detected and recorded in such a way that the relative intensity could be correlated with the distance along the plate. Samples were then punched from the plate at selected locations and analyzed for U^{235} content. By this technique, burnup (% original U^{235} atoms fissioned) values were obtained along the length of the fuel plate. The results have been plotted as burnup contours at 2%

intervals (see Figures 31 through 34). Four elements, listed in Table IV (two MTR and two ETR) have been scanned in this fashion.

TABLE IV

DATA ON FUEL ELEMENTS USED FOR GAMMA SCANNING

	Element	Reactor Position	MWD	% Core Power	Original Gm U235	Enrichment	Gm B
ETR	U620	M-8	2236	2.8	399.7	93.2	2.26
ETR	U635	M-10	2236	2.7	401.1	93.2	2.26
MTR	C419	31	717	3.0	200.8	93.1	
		33	225	4.6			
		16	107	4.4			
MTR	C423	33	717	4.8	200.1	93.1	

The relative gamma intensities were measured after calibrating the scanning device with a Cs-137 source so that a characteristic intensity drop could be detected at a counting rate of 660 counts/min. This was accomplished by adjusting the fine voltage. The discriminator was then set on 700, meaning that all intensities above 0.7 MEV were measured. Range control required adjusting from one element to another in order to keep the scans on the chart.

A photographic reduction of a typical scan is shown in Figure 35. The relative intensity values were then plotted against the % burnups, obtained by analysis, for each fuel element (Figure 36). At least six analytic specimens were taken from each element covering the range of relative intensity. From this relation between relative intensity and burnup, the burnup was determined for each 2% interval along a particular scan. Three scans were taken longitudinally, at center line and one inch on each side of the center line. Two percent isoburnout lines were then plotted.

Figure 31 a. Burnout Pattern of ETR Fuel Element U-620 Plates

Figure 31 b. Burnout Pattern of ETR Fuel Element U-620 Plates

Figure 31 c. Burnout Pattern of ETR Fuel Element U-620 Plates.

Figure 31 d. Burnout Pattern of ETR Fuel Element U-620 Plates

Figure 32 a. Burnout Pattern of ETR Fuel Element U-635 Plates.

Figure 32 b. Burnout Pattern of ETR Fuel Element U-635 Plates

Figure 32 c. Burnout Pattern of ETR Fuel Element U-635 Plates.

Contour Numbers Show % Of U-235 Fissioned.

Figure 32 d. Burnout Pattern of ETR Fuel Element U-635 Plates.

Figure 33 a. Burnout Pattern of MTR Fuel Element C-419 Plates.

Contour Numbers Show % Of U-235 Fissioned

Figure 33 b. Burnout Pattern of MTR Fuel Element C-419 Plates.

Figure 33 c. Burnout Pattern of MTR Fuel Element C-419 Plates.

Figure 33 d. Burnout Pattern of MTR Fuel Element C-419 Plates.

Figure 34 a. Burnout Pattern of MTR Fuel Element C-423 Plates.

Figure 34 b. Burnout Pattern of MTR Fuel Element C-423 Plates.

Figure 34 c. Burnout Pattern of MTR Fuel Element C-423 Plates.

Figure 34 d. Burnout Pattern of MTR Fuel Element C-423 Plates.

Figure 35 Typical Gamma Scan of Fuel Plate from Spent Element

Figure 36 Percent Burnup vs Relative Intensity for Four Spent Fuel Elements

4. Corrosion

a. Sample Plates

In the last annual report⁽¹⁾ a number of pictures were presented showing the appearance of sample fuel plates clad with a variety of aluminum base cladding materials. In general, plates clad with 1100 aluminum showed the least amount of change. Those clad with 6061 aluminum alloy became very dark when first exposed to the reactor environment, but gradually lightened in color with long exposures. This was also true to a slightly lesser extent for X8001, X8002, X8003, and 5052 aluminum alloys. APM-M-257 aluminum powder metallurgy clad fuel plates, however, became very mottled in appearance.

In order to obtain a better understanding of this phenomenon, additional testing has been done during the year.

Unfuelled sample cladding plates have been irradiated in the MTR in the same manner as the regular sample fuel plates. These have been photographed after every reactor cycle using the same procedures as were used to photograph the fuel plates. Figure 37 is a picture of these plates.

The appearance of the irradiated cladding plates is similar to that of the irradiated fuel bearing plates. The outline of the "core area" of the plate, of course, is missing. One set of coupons of APM-M-257, 6061, and X8001 Al were tested out of pile for three weeks in boiling, aerated, de-ionized water of the same purity and pH as that used in the MTR. No discoloration was produced, and no weight change was noted. Additional testing of this type is in progress; however, it would seem that the change in surface appearance of the fuel plates is an effect of radiation.

b. Corrosion Film Formation

Because of the interest developing in possible film formation on ATR fuel elements which is purported to increase the film temperature as much as 150°F, a similar question was raised about the ETR fuel element. The hot spot temperature in the ETR is only a few degrees (394°F compared to 424°F) below the expected ATR temperature (without film formation) and with the higher pH might result in formation of a corrosion film.

The surface of an outside plate from an ETR fuel element (U-620) was examined to determine the extent and thickness of the corrosion product. Three metallography specimens were cut from the "hottest" portions of this plate which was the "hottest" plate in the element as determined by gamma scanning (See Section II B 3). These specimens provided a total of 4.5 linear inches of plate surface viewing. They exhibited less than 0.03 linear inches of corrosion product with a maximum thickness of 0.0001 inch. The increase in surface temperature, assuming the corrosion product was boehmite and has a thermal conductivity of 1.5 Btu/hr-ft²-F°/ft, due to a film thickness of 0.0001 inch was calculated to be from 5° to 6°F.

8 x 10²⁰ nvt (th)

20 x 10²⁰ nvt (th)

35 x 10²⁰ nvt (th)

(A) APM M-257 Al

(B) 6061 Al

(C) 5052 Al

- Figure 37 -

Sample Cladding Plates Which Have Been Irradiated for Various Exposures.
Note: These pictures were made in the MTR canal through 7 feet of water.

8×10^{20} nvt (th)

20×10^{20} nvt (th)

35×10^{20} nvt (th)

(D) X8001 Al

(E) X8003 Al

Figure 37 (Con't)

5. Failed Fuel Plates

a. History

In the last year, 14 sample fuel plates have been removed from the L-piece due to suspected ruptures. Six of these plates had APM aluminum alloys (M-257, M-470, M-583) for cladding and picture frames. Of the plates having 1100 aluminum picture frames, two were clad with 5052 and one each with M-257, 6061 and X8002. Two had both 6061 picture frame and cladding and the last had 6061 cladding on an M-257 picture frame. The majority of ruptures appeared due to separation between the cladding and picture frame, Figure 38. The irradiation time ranged from 2-22 cycles with U-235 burnups from 11 to 82%.

Steadily increasing activity in the MTR cooling water led to the discovery of most of the suspected ruptures. Routine visual examination of the plates has not proved too satisfactory for detecting clad-picture frame separation on plate edges. This led to an inter-cycle test based on a method developed by Osborne⁽¹²⁾ of placing sample fuel plates in a water-filled aluminum tube closed at one end. Samples of the water are analyzed for I-132 (a daughter of fission-produced Te-132). When a positive response is found, closer visual examination usually reveals the rupture as can be seen in Figure 38. However, a recent report by Beaver, et.al., of ORNL⁽¹³⁾ has shown that this high water activity could come from uranium contamination of the fuel plate surfaces.

b. Metallurgical Examination

Ten of the previously mentioned failed plates (Table V) were given a detailed examination, i.e., close visual examination, ultrasonic inspection, bend testing, metallography, and hardness tests.*

The ultrasonic inspection was performed at the MTR canal using the equipment shown in Figure 39. Typical traces produced by this equipment can be seen in Figure 40. As seen here on some of the plates clear indications of defective areas were revealed while on other plates the indications were much less positive. However, these scans were useful in determining the section from which metallographic samples should be cut.

Plate number 63-10 which was clad with 5052 aluminum had a long separation between clad and picture frame as seen in Figure 41.

* Results of hardness tests were not completed at the time of this report.

Item 53N-8. 19 wt %, fully-enriched U-Al Alloy. 6061 P.F., 6061 Clad. 22% U-235 Burnup.

Item 54-9. 35 wt %, Fully-Enriched U-Al Alloy. M-257 P. F., 6061 Clad. 12% U-235 Burnup.

Item 79-3. 50 wt %, Fully-Enriched U-Al Alloy (3 wt % Sn) M-257 P. F. M-583 Clad. 18 % U-235 Burnup.

Item 57-5. 35 wt %, Fully-Enriched U-Al Alloy (1 wt % Si) M-257 P. F. M-257 Clad. 29% U-235 Burnup.

Figure 38. Ruptured Sample Fuel Plates Showing Separation Between Picture Frame and Cladding.

TABLE V

THERMAL NEUTRON RADIATION RECEIVED AND BEND TEST RESULTS FOR FAILED FUEL PLATES

Item	Core Composition	Aluminum Picture Frame	Aluminum Cladding	Thermal NVT x10 ⁻²⁰	Degree of Bend*	Photographs
63-10	18 w/o U in Al	1100	5052	11.8	180°	Figures 48 - 54
54-9	35 w/o U in Al with 1 w/o Si	M 257	6061	1.8	20°	Figures 65 - 69
76-5	18 w/o U in Al	M 470	M 470	6.2	30°	Figures 75 - 77
53 N-6	19 w/o U in Al	6061	6061	16.2	60°	Figures 60 - 64
61-14	18 w/o U in Al	1100	X8002	4.4	30°	Figures 57 - 59
57-5	35 w/o U in Al	M 257	M 257	4.9	< 10°	Figures 70 - 74
53-9	19 w/o U in Al	1100	1100	7.1	30°	Figures 55 - 56
74-20	50 w/o U in Al with 3 w/o Sn	M 257	M 257	5.8	25°	Figures 78 - 80
56-4	18 w/o U in Al	1100	M 257	24.8	< 5°	Figures 81 - 83
79-2	50 w/o U in Al	M 257	M 582	3.9	< 10°	

*Degree of bending at fracture.

Ultrasonic Scanning Equipment Used for Inspection of
Irradiated Fuel Plates

Figure 39

Plate No. 53 N-6

Plate No. 54 - 9

Plate No. 56 - 4

Plate No. 63-10

Typical Ultrasonic Scan Traces of Irradiated Plates

Figure 40

Figure 42 is a photomicrograph of a specimen taken from a "good" portion of Plate 63-10 showing no signs of lack of bond. Figure 43 is a cross section showing the end and location of the crack seen in Figure 41. Figure 44 shows the normal appearing microstructure of the fuel at high magnification. Although Figure 42 indicated bonding at the location, the cladding separated from the core when a bend test specimen was sheared, Figure 45, and the fuel fell out. The bend test (which consisted of bending a 1/2 inch wide transverse strip of the plate in a vise and measuring the degree of bend when fracturing occurred) showed high degree of bending since only the cladding was involved, but indicates the cladding had considerable ductility. The facts that blistering had not accompanied this failure, and that the core and clad were separated in the upper half of the plate (as indicated by plate dimensions in Table VI) and that complete separation was effected by shearing, all indicate a very low bond strength for item 63-10.

Item 53-9 shown in Figure 46 had some blistering before the separation indicating a better bond strength than Item 63-10 and indicating considerable pressure build up in a void, before separation. Figure 47 is a section off to the side of the blister showing the separation propagated along the core clad interface, and also showing voids the size of fuel particles which are evidence of core failure and likely origin sites for a blister.

M-938

Figure 41

Clad-Pictureframe Separation of Plate 63-10

Fuel Plate Thickness Measurement Orientation

PLATE NO°	DIMENSION		PLATE NO°	DIMENSION	
53N-6	A.	.052	74-20	A.	.032
	B.	.052		B.	.032
	C.	.065		C.	.032
	D.	.053		D.	.032
	E.	.052		E.	.032
76-5	A.	.050	79-2	A.	.0325
	B.	.050		B.	.0325
	C.	.050		C.	.033
	D.	.050		D.	.033
	E.	.050		E.	.033
61-14	A.	.052	63-10	A.	.060
	B.	.052		B.	.060
	C.	.052		C.	.051
	D.	.052		D.	.0515
	E.	.052		E.	.0615
53-9	A.	.052	54-9	A.	.037
	B.	.052		B.	.037
	C.	.052		C.	.037
	D.	.052		D.	.037
	E.	.052		E.	.037
57-5	A.	.056	56-4	A.	.053
	B.	.056		B.	.053
	C.	.056		C.	.053
	D.	.056		D.	.053
	E.	.056		E.	.053

TABLE VI FAILED SAMPLE FUEL PLATE
THICKNESS MEASUREMENTS

As Polished
Figure 42. Photomicrograph of a Specimen Punched from the Defective Portion of Plate 63-10
M-940

50 X
Figure 43. Photomicrograph Showing the End and Location of Crack
seen in Figure 41.

HF Etch 500 X
 Figure 44 Photomicrograph of 63-10 Showing Normal Microstructure
 of the Fuel.
 M-942

Figure 45. Plate 63-10 After Shearing, the Fuel Fell Out.

Figure 46. Plate 53-9 Showing Surface Crack and Slight Blistering of Edge

M-944

As Polished
Figure 47. Plate 53-9, a Section off to the Side of the Blister Showing the Location of the Crack and Particle Size Voids.

50 X

Metallographic examination of Item 61-14, revealed the peeled section (Figure 48) to be a core clad separation at a thick-core thinly-clad area (Figure 49). The cladding material is X8002 Al. With the exception of a few large fuel particles, Figure 50, the microstructure was normal for an 18 w/o U plate.

Considerable swelling occurred in Item 53 N-6 (Figure 51, 52, and 53) before separation between clad and picture frame progressed to the edge of the plate, indicating gas buildup at a void. Figure 52 shows voids adjacent to the location of maximum swelling (origin of failure); this is at the core run out. Figure 54 shows a normal microstructure at a sound portion of the plate. This plate was unusual because a thin strip of nickel was placed between the core and the clad to improve bonding. The fact that the propagation of the failure crack followed an irregular path through the "nickel" bonded area (Figure 55) indicates bond strength was improved. There was no indication of void around the nickel foil sections.

M-945

Figure 48

Plate 61-14 Has the Appearance of Peeling.

M-946

As Polished

50 X

Figure 49 Plate 61-14 a Section Through the Peeled Area Showing Core Thickening.

M-947

As Polished

500 X

Figure 50 Microstructure of Plate 61-14.

Figure 51 Plate 53 N-6 Showing Edge Defect

M-949

Figure 52 Polished Section Through Defect Area. Blistering and Voids in the Core.

50 X

M-950

As Polished
Figure 53 Plate 53 N-6 Showing Separation of Picture Frame and
Clad in Defect Area.

50 X

M-951

HF Etch
Figure 54 Microstructure of Plate 53 N-6 core

500 X

50 X

Figure 55 Polished Section Showing Location of Crack in the Plate Adjacent to the Blister. The crack Followed An Irregular Path.

Items 54-9 and 57-5 had similar type failures; both were 35 w/o U in aluminum and both formed large voids throughout the core. The swelling in Item 54-9 is seen in Figures 56, 57, and 58. The voids which formed in the core can be seen in Figure 59 (adjacent to blister) and in Figure 60, a section from a "sound" portion of the plate. Figure 57 also shows a void in the cladding caused by the diffusion and collection of gas. It is interesting to note, in Figure 59, that the voids are forming within the fuel particle.

The failure of 57-5 is similar to the previous one as seen in Figures 61, 62, and 63 with the exception that the voids in the area of the blister have formed preferentially in the matrix (rather than in the fuel particle), Figure 64, and in a "sound" area, Figure 65, voids formed preferentially in the area of the core-clad interface.

Items 76-5, 74-20 and 56-4 were similar, in that nothing could be found in the metallography to indicate cause of failure. The dimensions, Table VII, indicate uniformity. A flaw on the edge of 76-5, Figure 66, was investigated, Figures 67 and 68, but no defects of failure magnitude were found. A similar examination is seen for Item 74-20 in Figures 69, 70, and 71 and for Item 56-4 in Figures 72, 73, and 74. Since ultrasonic examination could find no defects, the most likely explanation for Items 76-5, 74-20, and 56-4 is the one contained in Reference 13 which discusses the diffusion of uranium to the surface of a fuel plate

Figure 56 Plate 54-9, Showing Edge Defect.

M-954

As Polished

50 X

Figure 57 Section Through Defective Area of Plate 54-9. Note That Voids Were Formed in the Cladding.

As Polished 50 X
 Figure 58 Plate 54-9 Showing the Blister Near the End of Core
 and the Crack Propagated Between the Clad and Picture
 Frame M-956

As Polished 500 X
 Figure 59 Core Adjacent to the Large Blister. Note Void Formation
 in the Particles.

M-957

As Polished
Figure 60 Plate 54-9 Showing a Section From the "Sound" Area
of the Plate.
50 X
M-958

Figure 61 Plate 57-5 Showing the Surface Defect.

As Polished 50 X
 Figure 62 Plate 57-5. A Section Showing Central Portion of
 Blister. M-960

As Polished 50 X
 Figure 63 Plate 57-5, Showing That Crack Propagated Through the
 Core

M-961

As Polished

500 X

Figure 64 Plate 57-5 Showing the Voids in the Core, Preferentially in the Matrix.

M-962

As Polished

50 X

Figure 65 Plate 57-5 Shows Voids at Core-Clad Interface

M-963

Figure 66 Plate 76-5 Showing Flaw on Edge.

M-964

As Polished 50 X
Figure 67 Plate 76-5 a Cross Section Showing Sound Core and Clad.

M-965

As Polished
Figure 68

Microstructure of Plate 76-5 Core

500 X
M-966

Figure 69

General Appearance of Plate 74-20

M-967

50 X

Figure 70 Polished Section of Plate 74-20

M-968

500 X

As Polished
Figure 71 Microstructure of Plate 74-20 Core

M-969

Figure 72

General Appearance of Plate 56-4

M-970

Figure 73

Polished Section of Plate 56-4

50 X

As Polished

500 X

Figure 74

Microstructure of Plate 56-4

during fabrication. If this should occur, coolant waters could be contaminated during irradiation even though the plates were smear clean before insertion in the reactor.

Since it was revealed by examining the burnout pattern on the plate, 79-2 and by X-rays that the core was unusually close to the edge of the plate, no further failure mechanism was sought for this plate.

c. Summary of Failed Fuel Plate Examination

To summarize the results of the examination, the evidence indicates the following:

1. 63-10 failed because of insufficient bond strength.
2. 61-14 had an area too thinly clad.
3. 53-9, 53 N-6, 54-9, and 57-5 were core failures in which voids enlarged and combined to form blisters and then cracking.
4. 76-5, 74-20, and 56-4 showed no signs of a material failure mechanism which would bring the coolant in contact with the core proper.

The uranium may have diffused to the surface of the cladding during fabrication. Simple "smear" testing for α contamination was not enough to detect this.

Because of these fuel plate failures, two additional pre-irradiation testing procedures have been established: (1) liquid nitrogen leak testing, and (2) whole plate α counting.

The first of these tests developed at Chalk River⁽¹⁴⁾, consists in immersing the plates in liquid nitrogen and following this with submerging the plates in alcohol. Any surface or edge defects are apparent from the stream of bubbles due to the boiling of previously absorbed liquid nitrogen. This test has given much better results than the fluorescent penetration, "zyglo", test which it has replaced.

In the second of these tests the whole plate is placed in the chamber of an α counter. Any significant amount of α contamination is cause for rejection. Using this procedure, plates with α contamination which would not "smear" off have been detected.

6. Reactivity Measurements

a. RMF

Periodic measurements in the RMF have continued to supplement the information obtained on various test plates. Previous progress reports (1&2) describe in detail the objectives and results of the RMF measurements. The additional measurements made are summarized in Table VII which lists the fuel and poison burnout for various type plates. These were the last measurements to be made on the RMF before it was shut down and dismantled in April, 1962. All reactivity measurements of this program were transferred to the Advanced Reactivity Measurement Facility (ARMF). Details of this work are given below.

b. ARMF

(1) Description

The ARMF is an improved version of the RMF. It is a pool-type low power reactor located in a separate canal in building MTR-660 east of the MTR building. A schematic of the core is shown in Figure 75. Structurally the ARMF is much more stable. This factor together with more advanced instrumentation has improved appreciably the reproducibility of the measurements. The degree of symmetry of the core is also greater. There are four cruciform safety rods in the corners and four lattice assemblies. Fuel elements are made up of flat instead of curved plates and are close packed. The positions of the coarse control shim and the servo regulating rod can be read with greater precision. The full scale regulating rod position indication is registered normally in 16 widths of the strip chart compared to 5 for the RMF. A reactivity change of $0.2 \mu k$ ($10^{-6} \Delta k/k$) can be read on the chart. Samples can be transferred between the ARMF and MTR canal through a hydraulic shuttle facility.

A new type lucite holder shown in Figure 76 has been designed and tested successfully in the ARMF water hole. This holder and the ARMF give reproducibility of measurements at least 5 times better than the RMF. A series of measurements were made on 6 plates with an average standard deviation of $.15 \mu k$.

(2) Measurements

The initial measurements made in the ARMF on the fuel plate development program were largely confined to calibration and to evaluation of reactor performance. Fig. 77 shows the fuel response curves for the two ARMF measurement positions indicated in Fig. 75. Plates made of 1100, 99.99%, and 99.997% aluminum were studied to determine the effect of residual boron, but no difference was observed. Further tests include MTR irradiation of these plates. These measurements are designed to improve the determination of poison content in these plates. A check of all standard plates showed good response and uniformity.

TABLE VII

RMF ESTIMATES OF FUEL CONTENT

Plate Number	Cumulative Radiation (nvt x 10 ⁻²⁰)	Pre-Irradiation Estimates		Post-Irradiation Estimates		% Burnout	
		U-235 (grams)	Poison Equivalent Boron (mg)	U-235 (grams)	Poison Equivalent Boron (mg)	U-235	Poison
38-2	1.8	*0.716	2.722	0.613	1.142	14.	58.
40-12	1.9	*0.698	3.725	0.620	2.951	11.	21.
41-6	2.2	*0.708	4.213	0.599	3.557	15.	16.
42-4	9.05	*0.707	3.857	0.358	.595	49.	85.
42-11	2.2	*0.707	3.857	0.582	1.102	18.	71.
47-4	5.82	0.760	1.070	0.509	1.025	33.	4.
48-3	6.54	0.736	8.610	0.471	1.307	36.	85.
48-5	2.0	*0.734	8.341	0.645	4.511	12.	46.
49-2	6.04	0.742	8.680	0.493	1.928	34.	78.
80-2	2.4	*1.050	12.725	0.904	7.272	14.	43.
* Based on unirradiated plates of same item.

Figure 75

ARMF CORE ARRANGEMENT

ARMF FUEL PLATE HOLDER

Figure 76

ARMF FUEL RESPONSE CURVES

Figure 77

(3) Discussion

The RMF results reported in Table VII indicate a reasonably linear relationship between irradiation exposure and fuel burnout. However, poison burnout appears to be scattered. This is probably due to the lower levels of poison concentration, different poison material, and possibly appreciable poison particle size. As indicated, it was necessary to make post-irradiation measurements for certain type samples on plates other than the ones on which pre-irradiation measurements were made. This also adds some uncertainty to the measurement since plates of the same type have significantly different initial reactivities. Improved sensitivity and precision are indicated in the ARMF measurements.

7. Flux Wire Considerations

The reliability of flux wires for determining U^{235} burnup has been evaluated. Burnup estimated from flux wires was compared with that determined from mass spectrographic analyses (95% confidence limits) for forty-six sample fuel plates. The differences between the two methods for determining burnups ranged from +43.2% to -24.5% with an absolute average of 12.5%. For the 27 plates with less than 40% burnup, the absolute average difference was 16.1%, while that of the 11 plates with greater than 60% burnup was only 6.3%.

As stated in IDO-16574, the flux wires are in areas of maximum flux and must be corrected by a flux perturbation factor. Because the L-piece is loaded differently each cycle, it is not practical to calculate a flux perturbation factor for each of the 96 plate positions and a constant value is used throughout the entire L-piece. As the above data indicates, the use of a constant perturbation factor results in large errors in burnup estimates.

In the future, flux wires will be used as a guide in estimating burnup with the ARMF or mass spectrograph furnishing the final burnup value.

III. GEOMETRY OPTIMIZATION

A. Improved MTR Shim Rod

A principle involving separately removable fuel and poison sections in shim rods for longer and more flexible service has been developed. A prototype of the new design, in the form of an MTR shim rod, has been tested in the MTR/ETR hydraulic facility. The MTR/ETR safeguards committee has reviewed the design and accepted it in principle, although no formal request for reactor insertion has been made. As a result of the hydraulic tests, several improvements in the design, mainly that of adding extra water passage slots in the upper and lower sections, were made. A question still persisted, however, with regard to the amount of flux peaking expected in the water gap between the poison and fuel insert sections.

Because of this, four PDQ-4 computer problems were run mocking up the shim rod in R-Z geometry; two represented a standard 132-gram MTR shim and two represented the proposed design with 150 grams of fuel. On the basis of the results from these problems, it is felt that the increased gap in the proposed design will raise the peak power generation in the shim fuel section about 3% and the average power about 1%.

This modified shim rod was also mocked up in the ETRC to determine the effect of the water gap. The data from this test substantiated the results from the computer problems quite well. Since this power increase is not a significant increase over the standard shim, it therefore appears that this design is suitable for reactor use.

B. 32 Plate Fuel Element

Recent advancements in aluminum research development have produced techniques for manufacturing high strength fuel plates. These techniques coupled with the more advanced high density fuels technology have made it possible to reduce the required fuel plate thickness thus producing fuel elements with higher thermal efficiencies, i.e., fuel elements that operate cooler and/or require less coolant at reactor power.

The MTR fuel element configuration has been optimized consistent with present technology, and two elements have been purchased for reactor use. These elements have 32-0.030" 6061-T6 clad fuel plates, separated by 0.068" water channels and contain a total of 250 grams U235. These elements provide a 68% increase in heat transfer area yet retain present metal/water ratio.

To permit testing these elements in reactor positions adjacent to standard MTR elements without increasing the lateral ΔP across the standard element external plate, the 32 plate elements have been constructed with a standard external channel (nominal 0.116" between elements).

The elements were hydraulically tested to 140% of reactor flow at 100°F (flow at 40 psi across the elements) and withstood the hydraulic forces produced by the unequal channels (there was actually a 0.0655" internal channel adjacent to the 0.120" external channel). The hydraulic

tests showed the average internal channel velocity to be 26.1 ft/sec with a variation of + 6% - 12%.

This element has the capability of a 46% increase in power or a 44% reduction in pumping rate (gpm) while maintaining the same maximum surface temperature.

These elements will be tested in the MTR as soon as AEC and Safe-guards approvals are obtained.

C. Concentric Rounded Square Fuel Element

A concentric rounded square fuel element has been designed.⁽²⁾ This fuel element consists of nine concentric square tubes with the corners rounded off and one cylindrical tube in the center. A "dummy" fuel element of this design, made of mild steel, has been tested in the hydraulic facility. Although a detailed analysis has not been made, a preliminary investigation of the data indicates that with a given ΔP across the fuel element, the total flow for the element is larger than for a standard ETR fuel element. The data indicate that the minimum velocity in any channel is approximately 30 ft/sec and in addition, the flow distribution across the fuel element appears to be relatively uniform.

It is anticipated that a detailed analysis will indicate the desirability of obtaining fully-fueled and operational prototypes to complete the testing.

D. Beaded Fuel Plates

One of the difficulties encountered when using aluminum clad fuel plates is, of course, the low mechanical strength of aluminum at elevated temperatures. These properties can be improved by alloying additions.⁽¹⁾ Another means of increasing the transverse stiffness of fuel plates is through the use of beads formed across the plate. Some of the variables to be considered in the design of these beads are the amplitude, spacing, and the shape to be used. It was believed to be desirable to have the amplitude no more than one-half the water channel thickness in the fuel elements so as to minimize the possibility of blocking the water channels. The shape chosen had to be such that there would be no danger of rupturing the core or cladding during fabrication. Figure 78 is a cross-sectional view of typical bead in a fuel plate showing the height and shape used.

The desired frequency with which the beads were spaced along the plate was determined to be a compromise between strength, resistance to fluid flow and the effects on heat transfer. It was desirable to achieve the maximum strength with the minimum effect on the other factors.

HF Etch

14 X

Cross Section of Beaded Fuel Plate

Figure 78

In order to determine the maximum spacing which could be used that would still give a minimum amount of deflection, a number of 2-foot long static pressure test sections were made. These used dummy fuel plates of MTR, ETR and ATR configurations. The plates on one side of the test sections were plain while those on the other side were beaded. Bead spacings of 6, 3, and 2 inches were tried.

The static pressure testing apparatus is shown in Figure 79. In these tests a vacuum was drawn on the test section until the unbeaded side took a slight permanent set. The section was next pressurized and the deflection readings recorded at pressure intervals of 1" Hg.

With the ribs spaced at 6-inch and 3-inch intervals, no improvement in the plate stiffness was noted. However, as can be seen in Figure 80 an MTR-type plate with a 2-inch bead spacing showed considerable improvement over an unbeaded plate.

From the results of this work, it can be concluded that the stiffness of fuel plates can be greatly improved by beads running across the fuel plate. The effect of these beads on the heat transfer and hydraulic characteristics of the fuel element have yet to be investigated.

Figure 79

Static Pressure Deflection Test of a MTR-Type Fuel Plate Test Section

Figure 80

Results of Static Pressure Testing of MTR Type Fuel Plates.

E. Pressure Deflection Testing of Fuel Elements

Three aluminum alloys have been static-pressure tested at room temperature to evaluate the strain predicted for fuel plates in a test reactor. The equipment used was that shown in Figure 79. The lateral pressure differential that produces the fuel plate loading and subsequent deflection

is developed by adjacent channels of unequal spacing. Each of the three alloys (6061-0, 1100-0, and X8001-0) was used to fabricate a non-fueled plate 0.080 in. x 3.9 in. x 2 ft., with a 5.4-in. radius of curvature. The 1100-0 and the X8001-0 plates were pinned, while the 6061-0 plates were roll-swagged into 2-1/2-in-wide side plates so that they formed the outside plates of a fuel section assembly. The side plates were supported at three axial positions (centered in the 2-1/2-in. dimensions) by bolts. For 0.250-in. side plates, this should give a fairly close approximation of the side plate deflection that would be encountered in actual practice while in the case of 0.150-in. side plates, the side plate deflection would be higher than the deflections in actual elements due to the lack of interior support plates.

Figures 81 and 82 show the stress-strain relationships for the various alloys and side plate thicknesses at room temperature. It is apparent that at room temperature any of the alloys could tolerate pressures of at least 10 to 15 in. Hg. across the plate. The deflections are in the direction to reduce the ΔP ; i.e., if at startup there was a pressure of 10 in. Hg. across it, the plate would deflect up to 0.010 in. or 0.030 in. depending on the alloy, side plate thickness, etc., and stabilize with a pressure of less than 10 in. Hg. acting across it. It would deflect until the elastic force equaled the ΔP driving force remaining. The plastic flow yielding and elastic deformation both would occur as the reactor flow was brought up to design values (prior to irradiation which would tend to reduce the elongation) and only the elastic component would be required as the reactor went up and down in flow for refueling, shutdowns, scrams, etc. Additional work is planned using test sections with 19 plates to be tested at elevated temperatures.

Figure 81 Total Deflection of Test Plates

Figure 82 Plastic Deflection of Test Plates

IV. RADIATION EFFECTS

Associated with the Reactor Fuels and Materials Development program are two projects: The in-pile high temperature testing of fatigue in steels, and of growth, swelling, and strength changes of beryllium. These projects are being undertaken to evaluate lifetimes of materials in reactors of higher fluxes and higher operating temperatures. They will be reported separately in a subsequent period.

The In-Pile Fatigue Project has progressed through the conceptual design. Preliminary design studies on the method of cycling (rotating-bending, or axial loading) indicate that an electro-hydraulic system may be the most desirable approach. The area of uncertainty lies in the resistance to irradiation. This is currently being investigated. Specifications for the electro-hydraulic actuator and all controls for the system are written. The steels selected for study include three carbon steels (A 212 Grade B, A 302 Grade B, A 350 LF-3); two stainless steels (type 304L, type 347); and two precipitation hardening stainless steels (AM 350, 15-7 MO). The time schedule calls for start of in-pile testing during Fy 64.

The Beryllium irradiation program which is designed to give information to the damage threshold when irradiated at high temperature is one phase of a larger program. A low temperature (ETR process water temperature) irradiation project is also being reported⁽¹⁵⁾ separately. The Beryllium irradiation program has progressed through the design of process and control equipment, and assembly work has begun on the instrument console.

V. CONCLUSION

The continuation of the Fuel and Materials Development program during the past year into studies of new materials, higher fuel loadings and prototypes leads to the following conclusions:

1. Uranium-aluminum alloys can be irradiated to long burnups under MTR conditions without dimensional distortion. This is true for a wide range of fuel densities up to 50 wt% alloys clad in any of a number of aluminum compositions.
2. When U-Al alloys are irradiated at temperatures up to 350°F no major changes are detected in the microstructure. Post irradiation annealing of these samples at 650°F does produce some micro changes, but dimensional stability is retained.
3. Both UO_2 and U_3O_8 react with aluminum under irradiation to form UAl_4 and Al_2O_3 (although the latter product has not been completely verified).
4. Irradiated fuel specimens exhibit increases in tensile strength and hardness and decreases in ductility even after short exposures. Samples with zero ductility were found over a wide range of compositions and radiation levels. The maximum elongation after irradiation was 4.5% in a one-inch gauge length compared to unirradiated elongations of 8-31%.
5. High weight percent fuels (e.g., 35 wt%) are feasible by a combination of alloy and powder metallurgy techniques. These retain the advantages of pure uranium-aluminum fuels yet permit fabrication of high fuel densities with uniform distributions. This use of the inter-metallic UAl_3 dispersed in powdered aluminum offers interesting potential in other areas of nuclear fuel materials.
6. Powder metallurgy continues to offer one of the best means of incorporating uniformly distributed burnable poisons in fuel cores.
7. Metallurgical bonding between cladding and core and between cladding and picture frame continues to be a problem with certain combinations of materials. This has been particularly true in attempting to bond aluminum to thorium, beryllium to aluminum and 6061 aluminum to itself. A technique for detecting minor cracking or separations which extend to the surface has been used successfully. This test which has been developed at Chalk River involves submersing the fuel plate in liquid nitrogen followed by immersion in alcohol.
8. Beryllium, because of its high scattering and low absorption cross sections, has been tested as a dispersant and as a cladding material for nuclear fuels. Further development is required, however, before the fabrication techniques are acceptable.
9. Attractive increases in strength are available from use of cold worked, or solution heat-treated and aged aluminum alloys; however, the effect of irradiation and temperature on these temper conditions

remains to be determined. Irradiation at temperatures up to 200°F apparently does not destroy the pre-irradiation hardness and strength.

10. Optimization of fuel plate strength, channel spacing, number of fuel plates and surface roughening offers considerable potential for increasing power density, reduction in pumping requirements or greater safety of operation. Heat transfer calculations and hydraulic tests on a 32-plate MTR element show that increases of 46% in power or 44% reduction in pumping rate are possible with this design.

11. "Smear" testing for α contamination on the surfaces of fuel plates is not satisfactory as it will not detect the contamination which results from the uranium in the core diffusing through the cladding. Consequently, a technique of α counting the whole sample fuel plate has been developed.

VI. FUTURE WORK

Future effort in this program will place more emphasis on materials development both in fabrication and in radiation testing. However, a limited effort will continue on prototype testing of interesting geometries in the MTR and ETR and in heat transfer and hydraulic studies on roughened fuel surfaces. The immediate objective of the materials effort will be to develop aluminum compositions capable of higher temperature operation. It is believed that 400-500°F operation is feasible with an extension of existing technology and that 600-800°F operation is achievable. Towards this end APM aluminums, iron-aluminum and beryllium-aluminum alloys will be studied in the HB-2 loop in the MTR. Higher additions of aluminum (>50-50) to the beryllium-aluminum system will be tested. Because of the use of beryllium as a structural member (moderator) as well as its possible use as a fuel cladding, some work will be continued on the determination of the damage threshold in this material under high temperature radiation. These tests will be performed in lead-type capsules involving sample temperature control up to 1470°F.

The work on tensile properties of the sandwich fuel plates will be extended to higher temperature. Pre-irradiation testing can now be accomplished at 400 and 600°F and equipment will be installed to duplicate these tests on irradiated samples in the Hot Cell.

Tensile tests and microhardness measurements will serve as the basis for further work on overaging of heat treated alloys and cold worked materials under irradiation and at elevated temperatures. While this problem is an acute one for the ATR fuel element, it is also important to existing reactors which use these materials as fuels or as structural members. Tests of several materials in different tempers will be irradiated at sufficient temperature levels to determine the temperature-exposure effect on their mechanical properties.

A joint effort with the Reactor Physics Research Group at Phillips will involve a study of the long term burnup of Thorium-U233 samples. The portion of work to be accomplished under the Fuels and Materials Program will involve the pre-post metallurgical inspection and evaluation of radiation stability through dimensional, chemical, and metallographic analyses.

Computer calculations performed in Fy62 using a one-dimensional code for determining fuel gradings necessary for flattening flux distributions will be continued and extended to two and three dimensional codes. As the data evolves from computer programs, critical experiments and in-pile tests on graded fuels will be performed.

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APPENDIX - DATA

TABLE VIII

ROOM TEMPERATURE TENSILE PROPERTIES OF IRRADIATED AND UNIRRADIATED FUEL PLATES

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\times 10^{-20}$ (1)	% Burnup U ²³⁵	Tensile Strength 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length (2)		
							Pre	Post	% Change	Pre	Post	% Change	Pre	Post	% Change
54 wt% UO ₂ , 20% enriched + Al cermet core 1100 Al clad	1	21	1.09	8	12.05	56									
		25	0.86	16	19.01	73		9.3			8.6			0.9	
18 wt% U, 93.5% enriched + Al alloy core 1100 Al clad	3	4	1.07	4	5.89	33									
		6	1.05	6	8.69	45									
22.7 wt% UO ₂ , 93.5% enriched + Al cermet core 1100 Al clad	4	15	0.88	3	3.66	23									
		18	1.04	8	11.48	55									
30 wt% UO ₂ , 93.5% enriched in Al cermet core 1100 Al clad	5	10	0.70	4	3.86	23									
		17	0.80	19	21.07	77									
45 wt% UO ₂ , 93.5% enriched in Al cermet core 1100 Al clad	6	24	0.59	3	2.45	15									
		22	0.90	7	8.66	45									
Upper $\frac{1}{2}$ 27.3 wt%, Lower $\frac{1}{2}$ 23.8 wt% U, 93.5% enriched + Al alloy core 1100 Al Clad	8	17	0.81	8	9.01	46									
		21	0.40	8	4.48	27									

(1) Plate numbers refer to irradiated plates only.

(2) The tensile specimen size was $\frac{1}{4}$ " wide x 1" gauge length.

TABLE VIII (Con't.)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U^{235}	Tensile Strength 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
							Pre	Post	%	Pre	Post	%	Pre	Post	%
							Pre	Post	Change	Pre	Post	Change	Pre	Post	Change
Upper $\frac{1}{2}$ 27.3 wt%, lower $\frac{1}{2}$ 23.8 wt% U+.08 wt% B, 93.5% enriched + Al alloy core, 1100 Al clad	10	1	0.80	7	7.75	41									
		18	0.80	8	8.83	45									
18 wt% U, 93.5% enriched + .057 wt% B in Al alloy core, 1100 Al clad	11	5	0.99	4	5.49	31									
		2	0.88	6	7.26	39									
18 wt% U 93.5% enriched in Al alloy core + .115 wt% B, 1100 Al clad	12	4	1.07	4	5.89	33									
		5	1.05	6	8.69	44									
22.7 wt% UO_2 , 93.5% enriched in Al matrix + .062 wt% B, 1100 Al clad	13	3	1.04	4	5.72	32									
		7	1.10	7	10.61	51									
22.7 wt% UO_2 , 93.5% enriched in Al matrix + .123 wt% B 1100 Al clad	14	4	1.01	4	5.57	31									
		5	1.11	5	7.64	40									
18 wt% U, 93.5% enriched in Al alloy + .0123 wt% Gd_2O_3 , 1100 Al clad	15	2	1.19	4	6.56	36									
		4	1.09	6	9.00	46									

TABLE VIII (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$	% Burnup U^{235}	Tensile Strengths 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
							Pre	Post	% Change	Pre	Post	% Change	Pre	Post	% Change
18 wt% U, 93.5% enriched + .057 wt% B + .013 wt% Gd_2O_3 in Al, 1100 Al clad	16	3	1.20	4	6.65	36									
		4	1.09	6	9.00	46									
22.7 wt% UO_2 , 93.5% enriched + 0.013 wt% Gd_2O_3 in Al cermet core + 1100 Al clad	17	5	0.87	3	3.62	22									
22.7 wt% UO_2 , 93.5% enriched + .062 wt% B + .013 wt% Gd_2O_3 + Al cermet core, 1100 Al clad	18	3	1.04	5	7.16	39									
18 wt% U, 93.5% enriched in Al alloy core 1100 Al clad	19	31	0.97	3	4.00	24									
		33	1.04	7	10.09	50									
100% ThO_2 core 1100 Al clad	20														

TABLE VIII (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U235	Tensile Strengths 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
							Pre	Post	% Change	Pre	Post	% Change	Pre	Post	% Change
98.5 wt% ThO ₂ + 1.5 wt% UO ₂ core, 1100 Al clad	21														
18 wt% U, 93.5% enriched + Al alloy core, 1100 Al clad	22	15	1.26	6	10.49	51									
		16	1.03	12	17.03	69									
		22	0.81	6	6.72	37	21.5	27.0	26	19.5	(3)		10	(3)	
18 wt% U, 93.5% enriched + Al alloy core 1100 Al + 0.5 wt% Ni clad	24	7	0.81	6	6.72	37	13.5	22.5	67	11.0	(4)		21	0	-100
		10	1.38	6	11.41	54									
		11	1.16	11	17.61	70									
		12	1.05	16	23.22	80	13.5	19.5	44	11.0	(4)		21	0	-100
18 wt% U, 93.5% enriched + Al alloy core 1100 Al + 1% Ni clad	25	7	1.29	6	10.66	52									
		9	0.99	19	25.99	83	16.5	17.0	3	8.0	(4)		23	0	-100
		12	0.81	5	6.72	37	16.5	26.5	61	8.0	(4)		23	0	-100
52 wt% UO ₂ , 30% enriched + Al cermet core 1100 Al clad	28	1	0.83	8	9.15	47	10.0	18.0	80	8.5	170	100	8	0.4	-95
		2	0.37	2	2.41	15	10.0	13.0	30	8.5	(3)		8	(4)	
42 wt% UO ₂ , 40% enriched + Al cermet core 1100 Al clad	29	1	0.83	8	9.20	47	11.5	18.5	61	9.0	16.0	78	13	0.8	-93.4
		2	0.57	3	2.35	15	11.5	9.5	-17	9.0	(4)		13	0.2	-98.5

(3) These values were not obtained because a cross-head extensiometer was used; this did not give correct elongation values.

(4) The ultimate load came before 0.2% offset.

TABLE VIII (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U ²³⁵	Tensile Strength 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
							Pre	Post	% Change	Pre	Post	% Change	Pre	Post	% Change
40 wt% UO ₂ , 60% enriched + Al cermet core, 1100 Al clad	30	1	0.83	8	9.20	47	11.0	16.0	45	5.5	(4)		14	0	-100
		4	0.42	4	2.30	15	11.0	15.5	41	5.5	13.0	136	14	1.8	-87.1
32 wt% UO ₂ , 80% enriched + Al cermet core 1100 Al clad	31	7	0.84	8	9.34	47									
		12	0.53	6	4.37	26	11.5	12.5	9	4.5	(4)		17	0	-100
35 wt% U, natural in Al alloy 1100 Al clad	32	6	1.31	10	18.05	71									
		9	0.83	8	9.20	47	15.0	24.5	63	10.5	19.5	86	7	1.5	78.6
		10	1.06	19	27.68	85	15.0	19.0	27	10.5	(3)		7	-2	
35 wt% U, 3% enriched + Al alloy core 6061 Al clad	33	6	0.63	12	10.52	51	22.0	26.5	20	14.5	(4)		6	0	
		7	0.65	5	4.47	26	22.0	39.0	77	14.5	30.5	110	6	2.5	-58
39 wt% UO ₂ , 20% enriched + Al cermet core 1100 Al clad	34	2	0.64	12	10.61	52	11.5	20.5	78	6.5	18.5	185	14	0.9	-93.6
		5	0.69	4	3.84	23	11.5	14.0	22	6.5	13.0	100	14	0.8	-94.3
65 wt% UO ₂ , 20% enriched + Al cermet core 1100 Al clad	35	3	0.63	12	10.55	52	8.5	10.0	18	6.0	(4)		12	0	-100
		7	0.67	4	3.72	23	8.5	12.0	41	6.0	11.0	83	12	1.1	-91

TABLE VIII (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux x 10 ⁻¹⁴	No. Cycles in MTR	Total Exposure nvt(th) x 10 ⁻²⁰	% Burnup U ²³⁵	Tensile Strength 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
							Pre	Post	% Change	Pre	Post	% Change	Pre	Post	% Change
18.7 wt% UO ₂ , 93.5 % enriched + .036 wt% B + Al cermet core 1100 Al clad	38	5	0.70	3	2.89	18	13.5	17.5	30	7.5	13.5	80	17	4.5	-74
		13	0.57	12	9.40	48	13.5	19.5	44	7.5	16.5	120	17	1.9	-89
18.7 wt% UO ₂ , 93.5% enriched + .951 wt% Dy ₂ O ₃ + Al cermet core 1100 Al clad	39	5	0.43	5	3.01	19									
		6	0.48	10	6.62	37									
		8	0.56	23	17.72	70	12.0	21.5	79	3.0	18.0	500	20	1.3	-94
18.7 wt% UO ₂ , 93.5% enriched + .037 wt% B, + .91 wt% Dy ₂ O ₃ + Al cermet core, 1100 Al clad	40	9	0.72	3	3.00	19									
		10	0.68	8	7.51	40									
		11	0.70	21	20.33	75	14.0	21.0	50	6.5	20.5	215	21	1.0	-95
18.7 wt% UO ₂ , 93.5% enriched + .036 wt% B + 1.275 wt% Ir + Al cermet core, 1100 Al clad	41	9	0.87	3	3.60	22	13.0	17.0	31	6.5	12.0	85	12	3.4	-72
		12	0.82	16	18.15	71	13.0	20.5	58	6.5	-		12		
18.7 wt% UO ₂ , 93.5% enriched + .0115 wt% Gd ₂ O ₃ + .036 wt% B + Al cermet core, 1100 Al clad	42	12	0.89	16	19.64	74	13.5	21.5	59	7.0	19.0	171	19	1.0	-95
		14	0.90	3	3.75	23	13.5	18.0	33	7.0	14.0	100	19	3.6	-81

TABLE VIII (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U ²³⁵	Tensile Strength 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
							Pre	Post	% Change	Pre	Post	% Change	Pre	Post	% Change
22.3 wt% U ₃ O ₈ , 93% enriched + Al cermet core, 1100 Al clad	47	10	0.99	3	4.13	25									
		11	0.95	8	10.57	52	9.5	21.0	121	4.5	-		6	0	-100
22.3 wt% U ₃ O ₈ , 93% enriched, .18 wt% B + Al cermet core, 1100 Al clad	48	8	0.98	3	4.05	24									
		11	0.95	8	10.55	52	10.0	18.5	85	4.5	-		5	0	-100
22.3 wt% U ₃ O ₈ , 93% enriched, .23 wt% B ₄ C + Al cermet core 1100 Al clad	49	9	0.80	3	3.30										
		11	0.82	8	9.10	47		19.5			-			0	
6.1 wt% UO ₂ , 93% enriched + Al cermet core 1100 Al clad	50	5	0.71	5	4.89	29									
		10	0.69	12	11.50	55									
11.6 wt% UO ₂ , 93% enriched + Al cermet core 1100 Al clad	51	6	0.84	6	6.98	38									
		14	0.79	16	17.42	70		21.0			-			-	

TABLE VIII (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$	% Burnup U^{235}	Tensile Strength 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
							Pre	Post	%	Pre	Post	%	Pre	Post	%
							Change			Change			Change		
16.8 wt% UO_2 , 93% enriched + Al cermet core, 1100 Al clad	52	6	0.63	6	5.20	30									
		8	0.66	11	9.99	50									
		10	0.62	21	17.95	71		18.5			17.0			1.1	
19 wt% U, 93% enriched + Al alloy core 6061 Al clad	53	27	0.81	6	6.72	37	24.5	28.0	14	15.0	-		16.0	0	-100
45 wt% U, 93% enriched + 3% Si + Al alloy core, 6061 Al clad	55	1	0.51	4	2.82	18	117	25.0	47	12.5	18.5	48	9	2.7	-70
19 wt% U, 93% enriched + Al alloy core APM M257 clad	56	1	0.76	3	3.16	20	31.0	42.0	35	27.0	-		3	0.2	-93
		9	0.81	6	6.72	37	31.0	36.5	18	27.0	-		3	0	-100
30 wt% U, 93% enriched in Al alloy core APM M257 clad	58	7	0.85	9	10.55	52		18.0			-			0	
45 wt% U, 93% enriched + 3% Si + Al alloy core, APM M257 clad	59	2	0.80	3	3.34	21		29.0			-			0	

TABLE VIII (Con't)

Plate Composition	Composition No. Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$	% Burnup U235	Tensile Strength 1000 Psi			Yield Strengths 1000 Psi			% Elongation in 1" Gauge Length		
						Pre	Post	% Change	Pre	Post	% Change	Pre	Post	% Change
32 wt% U, 93% enriched in Al alloy core X8001 Al clad	60	4	0.77	2	2.14	14	20.5	21.0	2	15.5	-	11	0	-100
		8	0.78	5	6.47	36	20.5	12.0	-41	15.5	-	11	0	-100
19 wt% U, 93% enriched in Al alloy core X8003 Al clad	62	4	0.78	6	6.47	36	22.0	39.0	77	16.5	-	6	0	-100
19 wt% U, 93% enriched in Al alloy core 5052 Al clad	63	1	0.74	2	2.04	13	27.5	33.0	20	21.0	-	8	-	
19 wt% U, 93% enriched + Al alloy core X8001 Al clad	73	5	0.78	6	6.47	36		26.0		-			-	

TABLE IX

MICROHARDNESS AND IRRADIATION DATA ON SAMPLE FUEL PLATES

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$ ⁽¹⁾	% Burnup U ²³⁵	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
54 wt% UO ₂ , 20% enriched + Al cermet core 1100 Al clad	1	✓ 21	1.09	8	12.05	56	28	44	57	36	250	590	550	360	- 35
		25	0.86	16	19.01	73		43	54		154	328		473	- 14
18 wt% U, 93.5% enriched + Al alloy core 1100 Al clad	3	✓ 4	1.07	4	5.89	33	22	34	54	33	150*	350	38	150*	290
		✓ 6	1.05	6	8.69	45		37	68		160*	380		160*	320
22.7 wt% UO ₂ , 93.5% enriched + Al cermet core 1100 Al clad	4	✓ 15	0.88	3	3.66	23	26	33	27	33	93	180	410	150	63
		✓ 18	1.04	8	11.48	55		35	35		110	230		410	
30 wt% UO ₂ , 93.5% enriched in Al cermet core 1100 Al clad	5	✓ 10	0.70	4	3.86	23	30	51	70	35	130	270	200	380	90
		17	0.80	19	21.07	77		58	93		145	414		-	
45 wt% UO ₂ , 93.5% enriched in Al cermet core 1100 Al clad	6	✓ 24	0.59	3	2.45	15	25	44	76	38	120	220	160	280	80
		✓ 22	0.90	7	8.66	45		83	230		230	510		320	100
Upper $\frac{1}{2}$ 27.3 wt%, Lower $\frac{1}{2}$ 23.8 wt% U, 93.5% enriched + Al alloy core 1100 Al Clad	8	17	0.81	8	9.01	46	34	38	11	32	119	240	200	237	18
		21	0.40	8	4.48	27		40	18		107	234		519	159

(1) Based on average 16-day cycle.

✓ Reported in past reports

* Particles too small for individual measurements - average values

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$	% Burnup U^{235}	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
18 wt% U, 93.5% enriched + .057 wt% B + .013 wt% Gd_2O_3 in Al, 1100 Al clad	16	✓3	1.20	4	6.65	36	18	34	89	29	125*	331	29	125*	331
		✓4	1.09	6	9.00	46		32	78		139*	495		139*	495
22.7 wt% UO_2 , 93.5% enriched + 0.013 wt% Gd_2O_3 in Al cermet core + 1100 Al clad	17	✓5	0.87	3	3.62	22	33	49	48	32	110	240	690	150	- 78
22.7 wt% UO_2 , 93.5% enriched + .062 wt% B + .013 wt% Gd_2O_3 + Al cermet core, 1100 Al clad	18	✓3	1.04	5	7.16	39	31	49	58	32	130	312	370	280	- 24
18 wt% U, 93.5% enriched in Al alloy core 1100 Al clad	19	✓31	0.97	3	4.00	24	22	26	18	26	120*	361	77	120*	56
		✓33	1.04	7	10.09	50		40	82		170*	550		170*	121
100% ThO_2 core 1100 Al clad	20														

* Particles too small for individual measurements - average values

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$	% Burnup U^{235}	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
Upper $\frac{1}{2}$ 27.3 wt%, lower $\frac{1}{2}$ 23.8 wt% U + .08 wt% B, 93.5% enriched + Al alloy core, 1100 Al clad	10	✓ 1	0.80	7	7.75	41	29	45	55	35 37	100	190	130	390	160
		18	0.80	8	8.83	45		30	3		56	87	150	228	52
18 wt% U, 93.5% enriched + .057 wt% B in Al alloy core, 1100 Al clad	11	✓ 5	0.99	4	5.49	31	26	43	65	37	160*	330	77	160*	110*
		✓ 2	0.88	6	7.26	39		34	31		130*	250		130*	170*
18 wt% U 93.5% enriched in Al alloy core + .115 wt% B, 1100 Al clad	12	✓ 4	1.07	4	5.89	33	22	35	59	33	129*	291	33	139*	291
		✓ 5	1.05	6	8.69	44		32	45		139*	321		139*	321
22.7 wt% UO_2 , 93.5% enriched in Al matrix + .062 wt% B, 1100 Al clad	13	✓ 3	1.04	4	5.72	32	23	37	61	32	108	237	445	249	- 44
		✓ 7	1.10	7	10.61	51		35	52		106	231		253	- 43
22.7 wt% UO_2 , 93.5% enriched in Al matrix + .123 wt% B 1100 Al clad	14	✓ 4	1.01	4	5.57	31	29	56	93	38	114	200	473	180	- 66
		✓ 5	1.11	5	7.64	40		50	72		148	299		430	- 9
18 wt% U, 93.5% enriched in Al alloy + .0123 wt% Gd_2O_3 , 1100 Al clad	15	✓ 2	1.19	4	6.56	36	19	31	63	25	141*	464	25	141*	464
		✓ 4	1.09	6	9.00	46		45	137		161*	544		161*	544

* Particles too small for individual measurements - average values

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$	% Burnup U235	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
98.5 wt% ThO ₂ + 1.5 wt% UO ₂ core, 1100 Al clad	21														
18 wt% U, 93.5% enriched + Al alloy core, 1100 Al clad	22	15	1.26	6	10.49	51	42	55	31	53*	180	240	53*	210	300
		16	1.03	12	17.03	69		72	71		200	280		240	350
		22	0.81	6	6.72	37		53	26		98	85		519	879
18 wt% U, 93.5% enriched + Al alloy core 1100 Al + 0.5 wt% Ni clad	24	7	0.81	6	6.72	37	33	37	11	43*	146	240	43*	-	-
		10	1.38	6	11.41	54		33	0		180	320		210	390
		11	1.16	11	17.61	70		51	55		210	390		240	460
		12	1.05	16	23.22	80		47	42		186	342		-	-
18 wt% U, 93.5% enriched + Al alloy core 1100 Al + 1% Ni clad	25	7	1.29	6	10.66	52	36	33	- 8	39*	300	670	39*	-	-
		9	0.99	19	25.99	83		47	31		191	516		-	-
		12	0.81	6	6.72	37		40	11		166	326		-	-
52 wt% UO ₂ , 30% enriched + Al cermet core 1100 Al clad	28	1	0.83	8	9.15	47	29	49	69	24	49	104	550	228	-59
		2	0.87	2	2.41	15		35	21		51	112		473	-14
42 wt% UO ₂ , 40% enriched + Al cermet core 1100 Al clad	29	1	0.83	8	9.20	47	29	43	48	33	56	70	520	-	-
		2	0.57	3	2.35	15		32	10		49	48		498	-12

* Particles too small for individual measurement - average values

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U ²³⁵	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
40 wt% UO ₂ , 60% enriched + Al cermet core, 1100 Al clad	30	1	0.83	8	9.20	47	27	50	85	30	81	170	410	228	-44
		4	0.42	4	2.30	15		39	44		62	106		383	- 7
32 wt% UO ₂ , 80% enriched + Al cermet core 1100 Al clad	31	7	0.84	8	9.34	47	26	44	69	27	96	256	320	102	-68
		12	0.53	6	4.37	26		48	85		69	155		-	
35 wt% U, natural in Al alloy 1100 Al clad	32	6	1.31	10	18.05	71	30	50	67	55*	76	38	55*	243	342
		9	0.83	8	9.20	47		50	67		73	33		228	315
		10	1.06	19	27.68	85		37	23		37	-33		67	22
35 wt% U, 3% enriched + Al alloy core 6061 Al clad	33	6	0.63	12	10.52	51	67	93	39	54	129	139	150	383	155
		7	0.65	5	4.47	26		83	24		119	120		276	84
39 wt% UO ₂ , 20% enriched + Al cermet core 1100 Al clad	34	2	0.64	12	10.61	52	31	42	35	34	49	44	440	-	-
		5	0.69	4	3.84	23		43	39		57	68		473	7
65 wt% UO ₂ , 20% enriched + Al cermet core 1100 Al clad	35	3	0.63	12	10.55	52	36	41	14	32	89	178	390	-	
		7	0.67	4	3.72	23		40			74	131			

* Particles too small for individual measurements - average values

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U^{235}	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
18.7 wt% UO_2 , 93.5 % enriched + .036 wt% B + Al cermet core 1100 Al clad	38	5	0.70	3	2.89	18	30	38	27	31	59	90	480	370	-23
		13	0.57	12	9.40	48		40	33		83	168		498	4
18.7 wt% UO_2 , 93.5% enriched + .951 wt% Dy_2O_3 + Al cermet core 1100 Al clad	39	✓5	0.43	5	3.01	19	30	38	27	32	54	69	380	670	76
		✓6	0.48	10	6.62	37		34	13		55	72		470	19
		8	0.56	23	17.72	70		47	57		61	91			
18.7 wt% UO_2 , 93.5% enriched + .037 wt% B, + .91 wt% Dy_2O_3 + Al cermet core, 1100 Al clad	40	✓9	0.72	3	3.00	19	32	34	6	37	59	59	450	-	
		✓10	0.68	8	7.51	40		39	22		62	68		440	- 2
		11	0.70	21	20.33	75		44	37		67	81		437	- 3
18.7 wt% UO_2 , 93.5% enriched + .036 wt% B + 1.275 wt% Ir + Al cermet core, 1100 Al clad	41	9	0.87	3	3.60	22	29	46	59	25	68	172	400	572	43
		12	0.82	16	18.15	71		52	71		80	250		401	0
18.7 wt% UO_2 , 93.5% enriched + .0115 wt% Gd_2O_3 + .036 wt% B + Al cermet core, 1100 Al clad	42	12	0.89	16	19.64	74	30	44	47	32	102	219	450	437	
		14	0.90	3	3.75	23		35	17		67	109			

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U ²³⁵	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
22.3 wt% U ₃ O ₈ , 93% enriched + Al cermet core, 1100 Al clad	47	✓ 10	0.99	3	4.13	25	26	35	35	30	150	400	140	-	-
		11	0.95	8	10.57	52		41	58		360	1100			
22.3 wt% U ₃ O ₈ , 93% enriched, .18 wt% B + Al cermet core, 1100 Al clad	48	✓ 8	0.98	3	4.05	24	30	37	23	29	130	350	178	-	-
		11	0.95	8	10.55	52		59	97		162	459			
22.3 wt% U ₃ O ₈ , 93% enriched, .23 wt% B ₄ C + Al cermet core 1100 Al clad	49	✓ 9	0.80	3	3.30		26	41	58	42*	140	230	42*	180	330
		11	0.82	8	9.10	47		43	65		143	240			
6.1 wt% UO ₂ , 93% enriched + Al cermet core 1100 Al clad	50	✓ 5	0.71	5	4.89	29	31	38	23	33	43	30	400	35	- 91
		✓ 10	0.69	12	11.50	55		31	0		40	21		-	-
11.6 wt% UO ₂ , 93% enriched + Al cermet core 1100 Al clad	51	✓ 6	0.84	6	6.98	38	47	32	-32	36	52	44	390	69	- 82
		14	0.79	16	17.42	70		35	-26		50	39		191	- 51

* Particles too small for individual measurements - average values

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure nvt(th) $\times 10^{-20}$	% Burnup U ²³⁵	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
16.8 wt% UO ₂ , 93% enriched + Al cermet core, 1100 Al clad	52	✓6	0.63	6	5.20	30	43	39	-10	36	43	19	450	200	- 56
		✓8	0.66	11	9.99	50		41	- 5		57	58		400	- 11
		10	0.62	21	17.95	71		44	2		58	61			
19 wt% U, 93% enriched + Al alloy core 6061 Al clad	53	27	0.81	6	6.72	37	50	41	-18	39	178	356	240	455	90
45 wt% U, 93% enriched + 3% Si + Al alloy core, 6061 Al clad	55	1	0.51	4	2.82	18	40	81	102	25	162	548	110	473	330
19 wt% U, 93% enriched + Al alloy core APM M257 clad	56	1	0.76	3	3.16	20	67	82	22	38*	121	218	38*	237	523
		9	0.81	6	6.72	37		96	43		188	395		437	1050
30 wt% U, 93% enriched in Al alloy core APM M257 clad	58	7	0.85	9	10.55	52	74	99	34	52	48	-8	220	634	188
45 wt% U, 93% enriched + 3% Si + Al alloy core, APM M257 clad	59	2	0.80	3	3.34	21	65	39	-40	36	56	56	140	401	186

* Particles too small for individual measurements - average values

TABLE IX (Con't)

Plate Composition	Composition No.	Plate No.	Average Thermal Neutron Flux $\times 10^{-14}$	No. Cycles in MTR	Total Exposure $\text{nvt(th)} \times 10^{-20}$	% Burnup U^{235}	Cladding Hardness			Matrix Hardness			Particle Hardness		
							Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change	Pre DPN	Post DPN	% Hardness Change
32 wt% U, 93% enriched in Al alloy core X8001 Al clad	60	4	0.77	2	2.14	14	29	36	24	48	66	37	74	437	491
		8	0.78	5	6.47	36		37	28		130	171		276	273
19 wt% U, 93% enriched in Al alloy core X8003 Al clad	62	4	0.78	6	6.47	36	46	60	30	86	156	81	100		
19 wt% U, 93% enriched in Al alloy core 5052 Al clad	63	1	0.74	2	2.04	13	57	80	36	45	133	196	140		
19 wt% U, 93% enriched + Al alloy core X8001 Al clad	73	5	0.78	6	6.47	36	36	45	25	47	48	2	120	455	279

TABLE X

PURPOSE AND STATUS OF PHASE IV AND V SAMPLE FUEL PLATES

Composition Number	Composition of Plate	Cladding Material	Cladding Thickness (Mils)	Core Thickness (Mils)	Purposes						Status	
					Fuel Stability	Fabrication Techniques	High Strength	Corrosion Resistance	Cladding Materials	Core Materials	Fabricated	Irradiated
74	50 w/o U, fully enriched, 3 w/o Sn in Al alloy APM-M-257 P.F.	APM M257 Al	12	8	X	X	X		X	X	X	X
75	50 w/o U, fully enriched, 3 w/o Sn in Al alloy APM-M-257 P.F.	M257* X8001 Al	12 5	8	X	X	X	X	X	X	X	X
76	19 w/o U, fully enriched, 1.5 w/o Si in Al alloy APM-M-470 P.F.	APM M470 Al	15	20		X	X		X	X	X	X
77	50 w/o U, fully enriched, 3 w/o Sn in Al alloy APM-M-257 P.F.	APM M-470 Al	12	8	X	X	X		X	X	X	X
78	18 w/o U, fully enriched, .8 Sn 1.5 Si in Al alloy APM-M-257 P.F.	APM M583 Al	15	20	X	X	X		X	X	X	X

* Duplex cladding

TABLE X (Con't)

Composition Number	Composition of Plate	Cladding Material	Cladding Thickness (Mils)	Core Thickness (Mils)	Purposes						Status	
					Fuel Stability	Fabrication Techniques	High Strength	Corrosion Resistance	Cladding Materials	Core Materials	Fabricated	Irradiated
79	50 w/o U, fully enriched, 3 w/o Sn in Al alloy APM-M-257 P.F.	APM M583 Al	12	8	X	X	X		X	X	X	X
80	19 w/o U, fully enriched in Al alloy. APM-M-257 P.F.	1100-0* Binal (3.5% B)	14 1	20		X			X	X	X	X
81	18 w/o U, fully enriched 0.4 w/o Si, 0.2 w/o Sn, 0.6 w/o Mg 0.2 w/o Cr in Al alloy Alclad 6061 Al P.F.	6061-T6 Al	15	20	X	X	X		X	X		
82	18 w/o U, fully enriched, 0.6 w/o Mn, 0.4 w/o Fe in Al alloy X8001-Al P.F.	X8001 Al	15	20	X	X		X	X	X		
87	32 w/o UO ₂ , fully enriched, 8 w/o BeO--Al cermet 1100 Al P.F.	1100 Al	15	20	X	X				X	X	

* Duplex cladding

TABLE X (Con't)

Composition Number	Composition of Plate	Cladding Material	Cladding Thickness (Mils)	Core Thickness	Purposes						Status	
					Fuel Stability	Fabrication Techniques	High Strength	Corrosion Resistance	Cladding Materials	Core Materials	Fabricated	Irradiated
88	30 w/o U, fully enriched in Al alloy. 1100-Al P.F.	Be	20	10		X		X	X			
89	18 w/o U, fully enriched in Al alloy 1100 Al P. F.	50/50 Be-Al Alloy	15	20		X		X	X			
90	Th plus 1.5 w/o U, fully enriched alloy, APM-M-257 P.F.	APM M257 Al	15	70	X					X		
91	63.5 w/o (ThO ₂ + 1.5 w/o UO ₂) in Al cermet. APM-M-257 P.F.	APM M257 Al	15	70	X	X				X	X	
92	47 w/o UC, fully enriched in Al cermet. APM-M-257 Al P.F.	APM M257 Al	20	8	X	X				X	X	
93	47 w/o UN, fully enriched in Al cermet APM-M-257 Al P.F.	APM M257 Al	20	8	X	X				X	X	

Table X (Con't)

Composition Number	Composition of Plate	Cladding Material	Cladding Thickness (Mils)	Core Thickness	Purposes						Status	
					Fuel Stability	Fabrication Techniques	High Strength	Corrosion Resistance	Cladding Materials	Core Materials	Fabricated	Irradiated
94	47 w/o U_3Si , fully enriched in Al Cermet, APM-M-257 Al P.F.	APM M257 Al	20	8	X	X				X		
113	50 w/o UAl_3 (69 w/o U, fully enriched) + .19 w/o B_4C + 6061 Al Powder cermet Al clad 6061 P.F.	X8001 Al	15	20	X	X	X		X	X	X	
114	50 w/o UAl_3 (69 w/o U, fully enriched) + .19 B_4C + X8001 Al Powder cermet X8001 Al P.F.	6061-T6	15	20	X	X		X	X	X	X	