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**QUARTERLY PROGRESS REPORT
METALLURGY DEVELOPMENT OPERATION
APRIL, MAY, JUNE, 1964**

THE STAFF OF METALLURGY DEVELOPMENT OPERATION

JULY 20, 1964

HANFORD LABORATORIES

**HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON**

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QUARTERLY PROGRESS REPORT

METALLURGY DEVELOPMENT OPERATION

APRIL, MAY, JUNE, 1964

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July 20, 1964

Hanford Laboratories

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

Work performed under Contract No. AT(45-1)-1350 between the
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Previous reports for Metallurgy Development Operation were:

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HW-77181	January, February, March, 1963
HW-78157	April, May, June, 1963
HW-78984	July, August, September, 1963
HW-80112	October, November, December, 1963
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QUARTERLY PROGRESS REPORT
METALLURGY DEVELOPMENT OPERATION
APRIL, MAY, JUNE, 1964

I. SUMMARY

METALLIC FUELS TECHNOLOGY

New Fuels Design

In the investigation of uranium alloys seven ingots have been consumable arc melted. These ingots will support investigation of variables including composition, melting program, and heat treatment. Composition variables include the use of additions of Fe, Si, Al, Mo, and/or Zr and variations in carbon content from approximately 50 to 1000 ppm.

Fabrication of a series of Zircaloy-2 clad fuel rods containing a series of dilute uranium alloys continued. Extrusion billets of 16 alloys are currently being prepared by melting, casting, and extruding to the desired billet size.

The dispersion of UO_2 in tungsten in the form of submicron particles is being studied by a coprecipitation technique. During the current period the product formed after calcining and reduction began to exhibit an apparent discoloration—the bulk of the material was olive drab in color rather than metallic gray. Investigations disclosed that nitrogen impurities were not the source of difficulty. Increase in reduction temperature or prereduction heat treatment at temperature above 1300 C produced metallic colored products. Observations to date indicate that the apparent discoloration of the as-reduced product is due to particle size rather than incomplete reduction or impurities.

Studies have been conducted to determine the effect of finely dispersed UO_2 on the tensile properties of iron at elevated temperature. The dispersions were achieved by coprecipitation, calcining, and reduction. Iron containing 1% UO_2 exhibited 15% increase in room temperature

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strength and iron containing 2% UO_2 , 35% increase. At 800 C, the strength of both the 1% and 2% UO_2 material was approximately double that of pure iron.

Fabrication Development

Over 200 coextrusions of C-64 Al clad, 0.6% Li-Al have been fabricated to provide stock for target elements. It is necessary to establish coextrusion conditions which would not bond the Li-Al alloy to the cladding but which would provide high quality cladding surfaces. These goals were met at extrusion ratios up to 20 to 1.

The electron beam vapor deposition process has been applied to the placement of copper on end caps for the projection welded and brazed end closure. A mask has been developed which limits the deposit to the areas to be brazed, thus making it possible to use use caps that are coined to final dimensions.

A third group of 20 hot headed N outer fuel components have been closed by projection welding on a 1200 KVA 600,000 ampere resistance welder. A new "phonograph finish" end cap was utilized on five pieces and appears to be very successful.

Tests have been devised to simulate the problem of removing damaged fuel elements from thin wall Zircaloy process tubes, to determine the force required to remove the fuel element, and to assess damage occurring in the process tube. These tests were used to evaluate the comparative performance of standard bridge rail aluminum supports versus a split weld arch rail design.

End closures brazing and thermocouple brazing have been completed on 12, thorium alloy, Zircaloy-2 clad fuel elements designed for use as heat transfer probes. These elements are now ready for irradiation.

Small diameter uranium wires were prepared to determine the strength of the material at sizes approaching fiber dimensions. The strength of bare uranium wires increased as the wire diameter was

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reduced until a peak value was reached at a diameter slightly less than 0.010 in. At this point ultimate strength at room temperature was approximately 275,000 psi. Strength of smaller wires dropped off rapidly with diameter probably because of the effect of inclusions on higher strength. Further tests should be made on ultra pure uranium to minimize effects of inclusions.

To evaluate the application of an asphalt lubricant to tungsten and the strength of drawn tungsten wire, 10 mil tungsten was cold drawn to 2.1 mils. The maximum strength achieved in the tungsten wire was approximately 540,000 psi.

Formation of thin metallic films by electron beam evaporation is under study. The initial tests were conducted on heated glass and metal substrates. Tightly bonded Zircaloy films 0.0004 in. thick and copper films 0.0005 in. thick have been formed. Factors affecting film quality are being evaluated.

Environmental testing of materials for the High Temperature Lattice Test Reactor continued. To date the only metals or alloys which have not been detrimentally changed by extended exposure at 1200 C to a graphite-nitrogen environment are commercially pure nickel and TD-nickel.

Metallographic examination of prototype N-Reactor fuels, after irradiation to exposures of 3000 Mwd/ton, have shown the primary cause of swelling in these test elements to be associated with grain boundary tearing rather than fission gas agglomeration.

The dimensional changes occurring in prototype N-fuel elements have been determined from pre- and postirradiation measurement data. Volume expansions computed from dimensional changes are in good agreement with those derived from bulk density measurements. Calibration data show that measurement equipment is accurate to within ± 0.0005 in.

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Irradiating testing of tubular fuel elements containing dilute uranium alloy fuels have shown that iron and aluminum additions decrease fuel swelling. Fuel containing 400 ppm iron and 800 ppm aluminum showed only approximately one-half the swelling observed in fuel elements of standard "N" fuel composition irradiated under identical conditions.

Irradiation of a large tubular fuel element having a fluted exterior surface continued successfully. A volume expansion of 2.2% has occurred after an exposure of 2400 Mwd/ton. Maximum fuel temperatures are calculated to be 550 C.

Initial irradiation testing of coproduct fuel element assemblies consisting of a tubular uranium fuel element with a central target core, has been completed. The irradiation performance of the uranium fuel tube was good. A maximum swelling of 3.3 vol% was observed at 1850 Mwd/ton exposure. No measurable swelling occurred in the target element.

METALLIC FUELS DEVELOPMENT

Irradiation of three tubular, Zircaloy-2 clad, Th-U fuel elements in a high-temperature, water-cooled loop continued successfully. An exposure of 0.9 at.% burnup has been achieved with less than 1% volume increase in the fuel.

Irradiation of metallic uranium fuel rods containing a submicron dispersion of uranium carbide is in progress. One sample has achieved a burnup of 0.3 at.% and irradiation of two additional capsules has been initiated.

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II. METALLIC FUELS TECHNOLOGY

NEW FUELS DESIGN

Uranium Alloy Investigations - R. G. Nelson

Seven ingots have been consumable arc melted in support of the above program. Variables include composition, heat treatment, and melting history. The uranium raw material is 1.47% enriched and is divided into dingot uranium (low carbon) and ingot uranium (500 ppm carbon). Heat No. 170, 171, and 172 (Table 2.1) were vacuum induction melted from the ingot material. These were extruded and remelted in the consumable arc furnace.

Heat No. 173 was double vacuum arc melted from the ingot uranium. Heat No. 174 was single vacuum arc melted and will be further diluted by remelting in the vacuum induction furnace. Heat No. 175 and 176 were arc melted from billet stock received from Mallinckrodt. Original plans called for this material to be heat treated and coextruded without arc melting; however, attempts to beta heat treat the billets produced severe cracking and salvaging of the material was necessary.

TABLE 2.1

ARC CAST URANIUM ALLOYS

Heat No.	Weight, lb	Composition, Aim						Remarks
		ppm				wt%		
		Fe	Si	Al	C	Mo	Zr	
170	68	400		800	500			Vacuum induction cast and arc cast remelt
171	55	175	85	100	1000			
172	65		800		500			
173	41						2.0	Double arc melted
174	58					6.0		Arc cast master alloy for vacuum induction remelt
175	55			800	Low			Cracked, Mallinckrodt billets, dingot, arc cast remelt
176	91		300	260	Low			

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Alternate Fuel Composition - J. W. Weber

Studies are in progress to determine the effects of altered fuel compositions upon fuel element fabrication, corrosion behavior, and irradiation performance. A part of this study involves the fabrication of uranium alloys into coextruded Zircaloy-2 clad rods, and their subsequent irradiation in NaK-filled capsules. Alloy additions to the uranium have been chosen from among those elements which are reported to impart improvements to the resistance to uranium fission gas swelling, grain boundary tearing, and corrosion.

Since the previous report period, two additional alloy compositions have been added, making a total of 16 compositions to be irradiated. The nominal compositions of these alloys and the alloys reported in the previous quarterly are shown in Table 2.2. To make the MA and MS alloys,

TABLE 2.2
COMPOSITIONS OF ALTERNATE URANIUM ALLOYS

<u>Alloy Designation</u>	<u>Fe</u>	<u>Si</u>	<u>Al</u>	<u>C</u>	<u>P</u>	<u>Zr</u>	<u>Nb</u>	<u>Mo</u>
R	150	100		500				
FAC	400		800	500				
FSC	150	100		1000				
A*	35	30	800	50				
s*	21	830	-	37				
SC	800			500				
SA*	24	290	260	37				
Z				500		2.0		
ZA*	64	25	700	30		1.75		
ZS*	55	80	-	17		1.55		
ZP*	45	20	10	18	55	2.25		
P*	20	32	-	41	450			
MA*	32	19	380	35			1.65	
MA			800	500				2.0
MS		350	-	500				0.1
SAC		400	800	500				

All compositions in ppm except Zr, Nb, and Mo, which are %.

* These compositions are averages of analysis received from Mallinckrodt Chemical Works.

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a master alloy of U-6 wt% Mo was vacuum arc-melted and cast into a 3 in. diameter ingot. Dilution of the master alloy and addition of the silicon and aluminum will be done during a vacuum-induction melting step. The SAC alloy will be produced by adding silicon and aluminum to a vacuum-induction melt heat. These three alloys will be cast into 4 in. diameter ingots and then α primary extruded to 2 in. diameter rods for coextrusion billets.

One billet each of the alloys FSE, SC, and FAC, which were initially prepared by vacuum-induction melting, were primary extruded to 2 in. diameter rods and machined to electrodes for vacuum arc melting. The arc-melted alloys, cast as 3.5 in. diameter ingots are now being prepared for α primary extrusion to 2 in. diameter rods for coextrusion billets.

The U-2 wt% Zr alloy has been double vacuum arc melted and is being prepared for α primary extrusion to a 2 in. rod for coextrusion billets.

Billets of two of the alloys, A and SA, which cracked on β heat treatment have been remelted in the vacuum arc furnace and are being prepared for α primary extrusion.

Submicron Dispersion of UO_2 in Tungsten - R. V. Bowersock

Under the sponsorship of the Lewis Research Center, NASA, a process is being developed to produce submicron-size UO_2 particles uniformly dispersed in a tungsten matrix. The approach under investigation is the coprecipitation of tungsten and uranium from solution, calcination of the precipitate to form the mixed oxides, and hydrogen reduction of the mixed oxides to produce a powder product containing 10, 20, and 30 vol% UO_2 . As reported last quarter, the feasibility of the process has been demonstrated; however, considerable difficulty was encountered this quarter in obtaining a metallic colored product. The difficulty manifested itself in that the product as reduced was olive drab in color rather than a metallic gray. Considerable evidence has been accumulated that indicate

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the problem to be only a color phenomenon associated with the extremely fine size of the inherent particles rather than incomplete reduction or impurities. Since a major portion of the effort this quarter was directed toward the solution of this problem a brief history of the problem as it developed is given as background information.

In our initial reduction work a metallic gray product was obtained both at 900 and 1000 C using standard bottle hydrogen. However, the product was not uniform in that segregation occurred during separation of the mixed precipitates from the supernate. Segregation was eliminated by modifying the chemical process to produce a more flocculent form of the tungsten precipitate. Details of the modification were given in last quarter's report. It was at this point that the as-reduced product began to exhibit an apparent discoloration, i.e., the bulk of the material was olive drab in color accompanied with a smut-like surface layer that was more pronounced toward the inlet gas end of the boat. Because of the apparent flow pattern of this surface layer it was assumed the difficulty was due to impurities being introduced with the reducing gas. The gas supply system comprising several feet of Tygon and surgical rubber tubing was replaced with an all stainless steel system and an ultrapure grade of hydrogen (10 ppm maximum impurities) was procured. When this upgrading of the reduction process failed to eliminate the discoloration, and testing of the exit gas during the early phase of reduction showed the gas to be basic, it was postulated that the problem was associated with residual nitrogen in the calcined product. To investigate the effect of higher temperature some of the discolored powder was heated in hydrogen to 1700 to 1800 C for 12 hr resulting in a metallic colored product. Metallographic examination of a compact made from the treated powder showed considerable agglomeration of the UO_2 had occurred. An alternate approach to the assumed nitride problem was to carry out the initial reduction in wet hydrogen to provide adequate moisture with which the nitrogen could react and be carried away in the exit gas. The initial reduction of one batch of

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mixed oxides was carried out at 750 C using hydrogen bubbled through room temperature water. The wet hydrogen reduction was continued until the tungsten trioxide had been converted to the dioxide; then, further reduction to the metal was accomplished using dry hydrogen at 1100 C until the dew-point of the exit gas had stabilized at a -50 F or lower. The as-reduced product still exhibited the olive drab discoloration. Kjeldahl analysis of the powder gave 30 ppm nitrogen as compared to 100 ppm nitrogen for powder reduced only in dry hydrogen. Portions of this wet hydrogen reduced powder were treated in vacuum at 1300 and 1500 C for 1 hr. The product at both temperatures had a metallic color. Compacts of the treated material are being prepared to determine the effect of the treatments upon UO_2 agglomeration. Summarizing the above observations one finds that in all cases where a metallic colored product was achieved there was associated an apparent increase in particle size resulting from either a coarser starting material as in the initial work or by partial sintering by use of higher temperature. Based upon • the above observation, • weight loss data, • no third phase observed in metallography, • wet chemical, spectrochemical, and X-ray diffraction analyses, it is concluded that the apparent discoloration of the as-reduced product is due to particle size and not a result of incomplete reduction or impurities.

A modification to the chemical process has been developed permitting larger batches to be processed with existing equipment. The change consists of using a slurry rather than a solution of ammonium paratungstate as the starting material. Because of the limited solubility of the tungsten salt, approximately 16 g/liter, only about 300 g batches could be produced by the old process. In addition to larger batches, the modified process eliminates both the laborious process of dissolution and clarification of the paratungstate and the reliance upon chemical analysis of individual batches for determining tungsten content of the starting material. The modified process has reduced chemical processing time by nearly one half. Four kilograms of coprecipitated mixed oxides containing

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20 vol% UO_2 have been produced by this slurry technique. This material will be reduced, then shipped to the Lewis Research Center for their evaluation.

To evaluate stringering and the effect of cold work and recrystallization upon the UO_2 particle size and distribution, a small billet was prepared by isostatic compaction at 80,000 psi. A compact density of about 75% of theoretical was achieved. The small billet was sealed within a heavy wall molybdenum can and extruded at a 9 to 1 ratio producing a dense, wrought product. Details of the extrusion process and the evaluation of the product are given in another section of this report.

Dispersion Strengthening of Iron with UO_2 - R. V. Bowersock and R. S. Kemper

Some work has been carried out to study the effect of finely dispersed UO_2 on the tensile properties of iron at elevated temperature. The UO_2 dispersion was achieved by coprecipitation of ferric hydroxide-ammonium diuranate by addition of ammonium hydroxide to an aqueous solution of the nitrate salts. The resulting flocculent precipitate was dried, calcined in air at 700 C, then reduced with dry hydrogen at 700 to 900 C. The as-reduced powders were isostatically compacted at 30,000 psi into right cylinders approximately 0.8 in. diameter by 2 in. long. The density increased from about 40% TD to 70 to 75% TD after compaction. Small billets were machined from the compacted cylinders and canned in a thick mild steel jacket. The billets were induction heated to 1050 to 1100 C, then extruded into a rod at a 9 to 1 reduction ratio. The iron data was obtained from pure iron powder prepared the same as the coprecipitated material.

Observations and conclusions derived from tensile data are:

- Addition of up to 2% UO_2 does not seriously reduce room temperature ductility
- All three materials exhibited yield point behavior in room temperature tests

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- The room temperature strength of the pure iron powder product is comparable with literature values for ingot iron.
- Increased room temperature strength was about 15% for the 1% UO_2 material and 35% for the 2% UO_2 material.
- At 800 C, the strength of both the 1% and 2% UO_2 material was approximately double that of pure iron.

Metallography of the as-extruded material at magnification up to 1500X revealed the following information:

- The pure iron powder had rather massive inclusions that stringered during extrusion, probably resulting from incomplete reduction.
- The 1% UO_2 and 2% UO_2 material had some massive inclusions but to a much lesser extent than the pure iron and the 2% material had less than the 1% material.
- Increasing UO_2 content caused reduction of grain size in the iron. Average grain size of the iron in the three materials was 0.026 mm for the pure iron, 0.019 mm for the 1% UO_2 material, and 0.013 mm for the 2% UO_2 material.
- Particle size of the UO_2 is 1 to 5 μ . This is an order of magnitude larger than previously coprecipitated material purchased by similar techniques.
- Distribution of the UO_2 appears fairly uniform and no voids were observed in the iron matrix.

FABRICATION DEVELOPMENT

Coextrusion of Aluminum Clad Target Elements - P. A. Ard

Over 200 coextrusion of C-64 aluminum clad, 0.6% Li-Al have been made to provide material for N-Reactor coproducer target elements. The target material is being clad with aluminum to provide a barrier between the Li-Al core material and the final Zircaloy-2 cladding. The billets were extruded to 0.500 in. OD rod at a 20 to 1 area reduction.

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The Li-Al cores were sealed within the aluminum cans by tungsten-inert arc welding and the composite billets were preheated to 175 C and extruded at a die exit speed of 600 in./min. These conditions were found to be adequate to insure unbond between the clad and core which is desirable for an additional diffusion barrier. Initial coextrusions extruded at 200 C and a die exit speed of 800 in./min (other conditions being the same) showed an occasional very light bonding in the lead portion of the extrusion.

The resulting surface finish on the coextrusions was in general very good. The use of an Aquadag (a colloidal graphite suspension) coating on the billets and a Bentonite grease containing lead fines (1/3 by volume) as a tool lubricant, drastically reduced the frequency of extrusion die pickup. The billets were prepared for the Aquadag coat by first vapor blasting the billet (240 grit size) which was found to be effective in holding the lubricant and preventing the aluminum from seizing to the die. The billets were then heated to 200 C and the Aquadag which was thinned with water at a ratio of 1 part Aquadag to 10 parts water was sprayed on the billets.

The core extrusion dies were prepared for extrusion by coating the die surfaces with a film of Electrofilm No. 1000 (a commercially available solid film lubricant). The coating was effective for about 12 to 20 extrusions after which the aluminum started to seize to the dies requiring the dies to be reconditioned with the solid film lubricant. A good oxide coating on the die seems to prevent die pickup, but appears less tenacious than the Electrofilm coating.

Work is being planned to coextrude aluminum clad Li-Al material from the 4 in. extrusion container at a reduction of about 36 to 1 in an effort to determine if bonding between the cladding core can be prevented at this reduction. Success of this test will enable the N-target material to be extruded from the 4 in. container on the product department extrusion press.

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Hot Headed Closure Studies - R. N. Johnson

A new and more economical process has been developed for the placement of copper on end caps used in the hot headed and projection welded end closure. The old process involved machining the caps, immersion plating with copper, and remachining, with a high reject rate on finished caps. The new process deposits a much higher quality and cleaner copper film by vacuum evaporation through a special mask, leaving the deposit only on the required areas of the cap. With this process it is possible to use caps that are coined to final dimensions, eliminating two machining steps and several delicate handling operations which are the cause of the current high reject rate on caps made by the old process.

In the new process, the copper was deposited on the heated substrate at the rate of 1 mil/hr at a vacuum of 10^{-6} Torr and a power level of 5 kw. Five-tenths mil (1.3×10^{-3} cm) of copper was deposited in the first run, but other thicknesses will be deposited to determine the optimum thickness for the closure. It is possible that 0.1 mil (2.54×10^{-4} cm) or less may be adequate.

Hot Headed Closure Welding - W. I. Steinkamp

A third group of 20 hot headed NPR outer fuel elements have been projection welded on the 1200 KVA 600,000 ampere resistance welder at the U.S. Army Rocky Mountain Arsenal in Denver, Colorado.

A new "phonograph finish" end cap was tried on five of these pieces. It appears to be very successful. In this end cap the projections are replaced by a 60° , 0.010 in. cut, 0.020 in. feed phonograph finish. Figure 2.1 shows a section of the closure using the phonograph finish cap.

Twelve of the pieces are undergoing 1500 psi 400 C steam autoclave testing. Three have been used for metallographic samples and five are being held for future testing.

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Note: Macro and micro cameras take reverse images which accounts for discrepancy between inset and photo.

Neg. 5Z4416F

13X

Neg. 5F4416E

50X

FIGURE 2.1

NPR Outer Fuel Tube, Hot Headed Projection Welded End Closure
with Phonograph Finish Cap
(Weld conditions on 1200 KVA, 600,000 amp, three-phase machine: 100%
phase shift; two impulses of six cycles; 1/2 cycle cool; weld force, 7000 lb;
forge force, 22,000 lb at beginning of second impulse)

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K-5 Reactor Tube Damage Tests (Preliminary Summary) - J. P. Pilger

Tests were conducted to assess fuel element support damage to K-5 process tubes in the event of a fuel element failure. Fuel element swelling was simulated by mechanically expanding a section of fuel cladding in a 16 in. length of process tubing. The cladding was expanded to give a predetermined load on the fuel supports. Standard bridge rails, and split weld arch rails were compared at loads 200 to 800 lb/support in 100 lb increments. Process tube deformation was measured with standard micrometer calipers and recorded. Extraction force versus distance was plotted by an X-Y recorder. Tube diameters were measured "over" the supports and "between" supports before deforming, at maximum deformation, and after extraction of the loaded section. Loads of 400 to 500 lb/support are necessary to produce any permanent set or change in process tube dimensions.

Summarized below are effects of rail load on process tube dimensions, and the break loose forces necessary to start the loaded section moving. The maximum extraction forces are also listed since this should be indicative of discharge forces in the event of fuel failure.

Process tube Damage

<u>Load Level/Rail, lb</u>	<u>Process Tube Diameter Change, in.</u>	
	<u>Maximum</u>	<u>Average</u>
200	0	0
300	0.001	0.0005
400	0.001	0.0009
500	0.0025	0.0016
600	0.006	0.0055
700	0.0065	0.0046
800	0.009	0.0064

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Discharge Forces

<u>Load Level/Rail, lb</u>	<u>Break Loose Force, lb</u>		<u>Maximum Force, lb</u>	
	<u>Bridge</u>	<u>Split Weld</u>	<u>Bridge</u>	<u>Split Weld</u>
200	230	100	520	190
300	290	40	550	180
400	450	-	800	-
500	440	500	1080	640
600	450	470	1100	910
700	640	690*	1240	780*
		550*		770*
800	660	640*	1320	640*

* Broken weld on one support

All tests were performed with the process tube in a vertical position, and no lubricant was used on the friction surfaces.

One factor favoring the split weld arch rail is the lower ultimate extraction force necessary to remove the sample from the tube section. Both supports have similar "break loose" characteristics, but the larger contact area of the bridge rail seizes much tighter after 3 to 4 in. of movement. This secondary seizing is apparently due to aluminum galling and buildup on the support surface. The smaller surface on the split weld configuration minimizes this condition. Visual examination of the process tube shows a larger aluminum deposit from the bridge rail, but in neither case does the process tube show any measurable wear.

Fabrication of Heat Transfer Probes - R. N. Johnson

End closure brazing and thermocouple brazing have been completed on 12, thorium alloy, Zircaloy-2 clad elements for heat transfer probes. Radiograph tests showed complete bonding and complete braze filling of the end closures with no detectable thorium contamination in all 24 closures. Three elements will not be used because of minor defects but all elements are being autoclave tested.

Uranium Wire Drawing - F. B. Quinlan

Uranium wire in a range of diameters has been prepared to determine the strength of the material at sizes approaching fiber dimensions. Uranium wire of unknown origin had been drawn a few years ago for testing the behavior of a wire-drawing lubricant.

For each mil that this wire was reduced a 4 in. sample was retained for future reference and a 6 in. sample was used for tensile tests. A curve has been plotted to show the effect of cold work and/or fibre size on the tensile strength of the uranium. In each case the wire was drawn through Carboloy dies using an asphalt base lubricant. This base coat was supplemented by MoS_2 powder. Zircaloy-2 clad uranium wire was also fabricated and tested. Coextruded Zircaloy-2 clad stock, approximately 3/8 in. in diameter with 20 mils of cladding, was swaged down to 47 mils in diameter. At this diameter it was given a 600 C stress relief anneal for 1 min. From this point on it was wire drawn to 17 mils in diameter and treated the same as the bare uranium wire. The bare uranium wire, at the start, had a 10.7 mil diameter. This wire was drawn down cold through 22 successive reductions to a final diameter of 3.75 mils. The original application of the asphalt lubricant sufficed for the entire 22 reductions; however, more MoS_2 was added at each step.

Photomicrographs were taken of transverse sections of the bare uranium wire at 500X. These studies show that the maximum particle size of the carbide, oxide, and nitride inclusions are $<5 \mu$ in the 10.7 mil diameter (Figure 2.2) starting material. However, micrographs of the type of extruded stock from which this wire was made, show that the inclusions (carbides) are usually $>30 \mu$. Attempts to resolve the grain size on the starting wire were unsuccessful; it must be extremely small. Figure 2.3 is a transverse section of the same wire after it was reduced to 4 mils in diameter. Again there appears to be noticeable reduction in the size of the inclusions, 2μ and less. Indications are that the inclusions

are friable and reduce in size as some function of the reduction ratio. This phenomenon should be more thoroughly investigated as these minute inclusions may possibly be made to add to, rather than detract from, the strength of the wire.

4Z4419A

FIGURE 2.2

Bare Uranium Wire,
10.7 mil Starting Diameter
(Increase in number of inclusions
per unit area)

500X

4Z4361A

FIGURE 2.3

Bare Uranium Wire,
4 mil Diameter
(Inclusions < 5 μ)

500X

The same effect appears in the larger Zircaloy-2 clad uranium wires (Figures 2.4a, 2.4b, and 2.4c). Figure 2.4a is a 500X transverse section of the starting material which is 47 mil diameter Zircaloy-2 clad uranium wire with a 600 C anneal for 1 min. There is little difference between the inclusion size in this sample and the one shown in Figure 2.4b which was

a. Diameter, 47 mils

b. Diameter, 26 mils

c. Diameter, 17 mils

Neg. 4Z4410D; 4Z4411D; 4Z4409D

FIGURE 2.4
Illustrating Breakup
of Inclusion Size
with Reduction in Diameter
(All 500X)

cold drawn from 47 to 26 mils in diameter but the difference does show up in the next reduction to 17 mils as shown in Figure 2.4c. It should now be noted that all of the photomicrographs show that the population density of the inclusions is inversely proportional to the diameter. This is to be expected as the fractured parts of the inclusions should move in a length-wise direction, during drawing, while the remaining parts of the original inclusions would retain their relative transverse positions. Figures 2.5a, b, and c are 100X views of Figures 2.4a, b, and c. These views show the increase in population density of the inclusions to a marked degree.

Figure 2.6 is a 100X view of the 47 mil starting stock after it was stress relief annealed at 600 C. The grain size of the uranium, which recrystallized during annealing, is 5 to 10 μ and the grain size of the Zircaloy-2 is too small to measure. This sample was prepared with the standard phosphoric acid, electrolytic etch and photographed under polarized light. In all subsequent reductions of this 47 mil stock it was impossible to again resolve the grain size; it became entirely too small, even at 750X.

Figures 2.7 and 2.8 are curves showing how the strength of these cold drawn uranium wires varied with the diameter. The strength peaks at slightly under 10 mils which represents approximately 99.6% cold work. Further tests should be made on ultrapure uranium, possibly starting with a large single crystal, which might extend this peak to much higher strengths at smaller diameters. It would also answer the question as to whether the inclusions are beneficial or detrimental to the wire strength.

5Z4410A

a.

4Z4411A

b.

4Z4409A c.

FIGURE 2.5

Illustrating the Increase in the Number of Inclusions
per Unit Area as the Wire Size is Reduced
(All 100X)

Neg. 5Z4410B

FIGURE 2.6

100X

Showing Grain Size
of 47 mil Diameter Starting Zircaloy-2 Clad Stock
(Grain size, 5 to 10 μ)

FIGURE 2.7

Cold Drawn Bare Uranium Wire,
(Strength versus Diameter)

FIGURE 2.8

Cold Drawn Zircaloy-2 Clad Uranium Wire
(Strength versus Diameter)

Extruded Tungsten Honeycomb Fuel - G. S. Allison

Previous extrusions of tungsten shapes, including honeycomb, have exhibited a geometry which is distorted by wrinkling of the surface of the tungsten. In all cases where this occurred either or both the molybdenum and tungsten billet components were fabricated from wrought stock. Since this did not occur in extrusions of as-sintered billet components, it is concluded that the wrinkling is caused by asymmetrical flow of extremely large grains which are formed in the wrought billet components during preheat. It was proposed that direct extrusion of as-compacted powder components would eliminate the wrinkling problem and also reduce the cost of the billet components. Several extrusions were made to evaluate this proposal. Extrusions of as-compacted (75% theoretical density) tungsten powder canned in mild steel and AISI 304 stainless steel preheated to 1200 to 1500 C were not successful. The gross mismatch between the can and core produced insufficient pressure to cause the tungsten to flow. The cores merely broke up into small pieces which were distributed along the steel extrusion. In another extrusion the core was a 75% TD tungsten powder cylinder containing seven equally spaced hexagonal molybdenum powder rods. It was made by vibrationally loading tungsten powder into a cylindrical rubber mold containing seven equally spaced hexagonal rods, withdrawing the rods and loading the resulting holes with molybdenum powder, and isostatically pressing to 75% TD. The compact retained the packed geometry and was canned in wrought molybdenum by vacuum electron beam welding. The entire assembly was held at about 600 C for 30 min. in vacuum immediately prior to sealing. After preheating from room temperature to 1850 C in 25 min it was partially extruded (25%) at 9 to 1 ratio before stalling at 200,000 psi. The geometry of the billet was retained in the extrusion and the W-Mo interfaces were exceptionally smooth, indicating that no excessive grain growth occurred during preheat. Metallography of a transverse section exhibited five recrystallized grains in both the molybdenum and tungsten, with a band of large grains in the external molybdenum rods parallel to the circumference. Density measurements show the

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tungsten to be 99.9% TD in the extruded portion and 98.3% TD in the unextruded portion. This experiment confirms that a dense, wrought, fine grained structure can be obtained from a direct powder extrusion and that the cross-sectional geometry of a complex composite powder billet is retained in such an extrusion.

One of the major stigmas attached to extruded honeycomb fuels has resulted from gross stringering of the UO_2 observed in extrusions of sintered W- UO_2 billets. It has been postulated that this stringering can be limited to an acceptable level by the use of billet fuel components in which the UO_2 is uniformly dispersed in the tungsten as submicron particles. To accomplish this it is necessary to start with the finest UO_2 particles obtainable and keep the overall processing time at temperature low to minimize migration and agglomeration of the UO_2 particles prior to extrusion. Coprecipitated W-10 wt% UO submicron powder was used in the initial extrusion experiment to investigate this idea. The powder (process described elsewhere in this report) was isostatically pressed at room temperature into a 75% TD compact 0.730 in. diameter by 2 in. long. This was sealed in a 2.125 in. diameter sintered molybdenum can by electron beam welding, preheated to 1875 C in 25 min, and extruded at 60 in./min ram speed and 9 to 1 ratio. The structure observed in a transverse section of the W- UO_2 is fine grained, 3 to 5 μ , tungsten containing the UO_2 as 1 to 3 μ irregularly shaped particles distributed at the grain boundaries. A longitudinal section exhibited elongation of the UO_2 particles to a length to diameter ratio of 9 to 1, with somewhat less elongation of the tungsten grains. The increase in size and relocation of the UO_2 particles to the tungsten grain boundaries apparently occurred during the 25 min billet preheat. Figure 2.9 shows microstructures at 100 and 1000X of etched transverse and longitudinal sections near the front of the extrusion.

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100X

Transverse

1000X

100X

Longitudinal

1000X

FIGURE 2.9

Etched Microstructure of W-10 wt% UO_2
Extruded at 9 to 1 Ratio
from As-Compacted Powder Preheated to 1875 C

Neg. 4Z4592C; 4Z4592D; 4Z4594C; 4Z4594D

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Fiber Strength Tests of Tungsten - F. B. Quinlan

Pure 10 mil tungsten thermocouple wire was cold drawn to 2.1 mils. The term "cold", in this case, refers to ambient temperature and not to any temperature below the brittle transition temperature. The wire was coated with asphalt base lubricant and MoS_2 . Nineteen reductions were made and the strength of each size was checked on the "Instron."

Figure 2.10 is a curve showing the results of this series of tests.

All of this work was done with carbide dies which did not constitute a complete uniform series of reductions. Such a set will soon be available and it is expected that strengths of the tungsten in future tests will approach 600,000 psi.

FIGURE 2.10
Cold Drawn Tungsten Wire

Electron Beam Evaporator - R. N. Johnson

Films of zirconium and copper have been deposited onto heated glass and metal substrates using the electron beam evaporator. These evaporations were part of a continuing program to determine the effect of substrate temperature and surface preparation on the quality and bonding of the metal films. Current evaporations of these metals have yielded high quality, tightly bonded zirconium films 0.4 mil (1×10^{-3} cm) thick and copper films 0.5 mil (1.3×10^{-3} cm) thick. Microscopic pinholes in the films have sometimes been a problem and appear to be correlated with surface preparation of the substrate. Other factors, such as rate of evaporation, distance from substrate, temperature, vacuum, and purity of evaporant have an effect on the quality of the film and are being studied. Instrumentation has been installed to accurately determine temperatures in various parts of the evaporator and substrate heater assembly.

Compatibility of Structural Materials with the HTLTR Environment -

D. P. O'Keefe

The evaluation testing of materials for the High Temperature Lattice Test Reactor includes fuel, cladding, control rod, thermocouple, and insulating materials. In summary, the test results to date have indicated that the only alloys that are not detrimentally changed by the reactor atmosphere, which is nitrogen with CO as an impurity, at 1200 C are TD-nickel and commercially pure nickel. These alloys will, however, react with some of the other materials to be used in the reactor in the presence of the reducing atmosphere. One such reaction is with the ceramic insulating brick, which interacts with nickel to form low melting compounds. Hastelloy B, which is relatively stable in the reactor atmosphere at temperatures below 1000 C, is resistant to attack by the refractory brick and may be substituted for nickel in the lower temperature regions of the reactor. The nickel alloys appear to react with some forms of B_4C to be used in control rods and with the UO_2 fuel in some circumstances. The compatibility of TD-nickel and nickel with the fuel and control

rod materials in the presence of the reactor atmosphere is the subject of current evaluation tests. Environmental tests are being prepared using B_4C in a graphite matrix in contact with TD-nickel, and graphite coated UO_2 in TD-nickel and nickel containers.

FUEL EVALUATION

"N" Fuel Assembly Tests - J. W. Goffard

Additional swelling and operational data have been obtained from two loop KER charges of prototype N-Reactor fuel elements (Tests K-4-15 and K-3-18). The two test charges were irradiated to tube average exposures of 1600 and 2000 Mwd/ton at prototypic temperatures and 1600 psi coolant pressure. Maximum fuel swelling observed in the 1600 exposure test was 0.85 vol% in an inner and 2.50 vol% in the corresponding outer fuel component. The maximum fuel swelling observed in the 2000 exposure test was 1.33 vol% in an inner and 3.64 vol% in an outer fuel component. Data from these two charges, shown in Figures 2.11 and 2.12, were used in the development of the empirical fuel swelling expression discussed in the previous quarterly. Also indicated in each figure are the calculated fuel swelling values for each component (dashed lines) obtained from the empirical fuel swelling expression.

Radiometallurgical examination has been completed on the highest exposure inner (NIE) and outer (NOE) components from the 2000 Mwd/ton exposure test. The inner component was irradiated to 2440 Mwd/ton and incurred 1.17 vol% fuel increase. The outer component was irradiated to 3070 Mwd/ton and incurred 3.64 vol% fuel increase. Superficially these components appeared to be in excellent condition without indications of warp, clad bumping or rippling, crud deposits, or corrosion effects. Metallographic examination of the fuel in the outer component (3.64 vol% ΔV) at 1000X reveals no obvious spherical porosity such as might be generated by fission gas agglomeration, but nonspherical voids are observed in colder areas of the fuel. These voids appear to be tearing

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FIGURE 2.11

Operational and Swelling Data
of NIE's from KER Tests
to 1600 (top) and 2000 Mwd/ton (bottom)

FIGURE 2.12

Operational and Swelling Data
of NOE's from KER Tests
to 1600 (top) and 2000 Mwd/ton (bottom)

at grain boundaries in the uranium fuel that is highly distorted during irradiation. Electron microscopy has given additional evidence of tearing and indicates that this form of porosity is the major contributor to the observed fuel swelling. Metallographic examination of the fuel in the inner component (1.17 vol% ΔV) at magnifications to 1000X did not reveal porosity. Electron microscopy does, however, reveal tearing in this fuel similar to that observed in the outer component, except that the voids are smaller in size. Both the microstructure of the fuel of the inner and outer components are shown in Figures 2.13 and 2.14.

Dimensional Performance of N-Fuels - V. L. Gelezunas and

L. E. Kuhlken

The initial analysis of N-fuel performance in the KER prototype irradiation tests has been completed and a document written.⁽¹⁾ The study included dimensional performance of the tube-in-tube N-fuels and an evaluation of the pre- and postirradiation measuring methods. The latter study will be useful in future in-reactor dimensional studies to be made after power ascension of N-reactor.

The swelling of the outer fuel elements calculated from pre- and postirradiation dimensional measurements was in good agreement with the values determined by changes in the bulk density. Swelling gradients were noted in individual elements. The data for the inner elements were not as reliable since the changes which occurred were of the order of the measuring precision.

The changes in the outside and inside radii of the fuel elements after irradiation following the swelling. The outside radius of the outers increased a maximum of 0.007 in., while the inside radius decreased a maximum of 0.004 in. A maximum change of 0.006 in. was measured in the outside radius of the inner fuel elements.

(1) V. L. Gelezunas. Dimensional Changes Observed as a Result of Irradiation of Prototype N-Reactor Fuel Elements, HW-83415. July 8, 1964.

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FIGURE 2.13

Micrographs of Fuel in NOE, 3070 Mwd/ton, 3.64% ΔV
(Top: 1000X, grain-boundary tearing porosity;
Bottom: 4500X, smaller porosity of tearing type)

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FIGURE 2.14

Micrographs of Fuel in NIE, 2440 Mwd/ton, 1.17% ΔV
(Top: 1000X, no resolvable porosity;
Bottom: 15,000X, porosity in hashed structure)

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The maximum "single-throw" warps (18.1 in. span) measured were 0.016 and 0.012 in. for the outers and inners, respectively. For the outers, the magnitude of the change in the warp vector was a strong function of the loop position. This did not appear to be the case for the inners.

Calibration between the pre- and postirradiation dimensional gages was accomplished by repeated measurement of a standard and several N-fuel elements. A "sag" which was found to occur in the postage was corrected for in the computer program which calculates the actual fuel dimensions from the raw gage data. The dimensions of all elements were compared statistically to determine to what extent there was a significant difference between the two machines.⁽¹⁾ The difference was shown to be negligible for the two gages for the measurement of outer elements. The number of inners measured to date is too small a sample size to provide significant error estimations. The reproducibility on both machines is within ± 0.0005 in. for all elements measured to date.

Testing of Alloyed Uranium Fuel - J. W. Goffard

Eight KSE-5 fuel elements were discharged from KER Loop 2 after attaining an average exposure of 1530 Mwd/ton. The loop charge consisted of alternately loaded 12 in. long fuel elements containing uranium with 120 ppm silicon and 140 ppm iron ("N" fuel composition) and elements containing 400 ppm iron and 800 ppm aluminum as additives. The purpose of the test was to compare the swelling performance of these two fuel materials at identical irradiation conditions in a high-temperature, high-pressure water reactor environment.

The volume mean fuel temperatures of these elements ranged from 425 to 525 C over the length of the charge. Fuel swelling for each element has been calculated from pre- and postirradiation bulk density measurements. The data shows a significant difference in the swelling behavior of the two fuel compositions with the Fe-Al additive fuel demonstrating

(1) J. L. Jaech. Components of Measurement Error at a Given Measurement Station, HW-73002. March 1962.

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consistently less swelling. Peak swelling of the Fe-Al containing fuel was 1.2 vol% compared to the peak swelling of the Fe-Si containing fuel of 2.1 vol%. Fuel swelling and exposure of the individual elements are portrayed graphically in Figure 2.15.

FIGURE 2.15

Graphic Portrayal of Exposure and Fuel Swelling Behavior of KSE-5 Fuel Elements Containing Two Fuel Compositions

Selected elements of this test charge will be submitted to Radio-metallurgy for postirradiation examination and evaluation. A similar charge of eight KSE-5 fuel elements of these two compositions, currently being irradiated in the KER facility to a goal of 2500 Mwd/ton exposure, will provide additional data on the effect of the Fe-Al additives on fuel swelling.

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Fluted Single Tube Fuel Elements - D. P. O'Keefe

The N-single tube fluted fuel elements (described in previous reports of this series) which are being irradiated in the ETR have completed 12 reactor cycles of irradiation in the high-pressure M-3 loop. Density changes in the elements have been monitored by underwater weight measurements after each reactor cycle. The average volume increase for the two elements throughout the exposure is shown in Figure 2.16. These elements are operating at a specific power level of 140 kw/ft with a corresponding maximum metal temperature of approximately 550 C. The irradiation is continuing.

FIGURE 2.16

Measured Volume Increase for Zircaloy-2 Clad Uranium
Fluted Fuel Elements Operating in 1600 psi Water

(The cross-sectional geometry of the fuel together with
the axial flow splitter is shown in the insert.)

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Coproduct Fuel Evaluation - J. W. Goffard

Thirteen 1.25% enriched driver elements and their companion Al-Li target elements were discharged from KER Loop 3 after attaining an average fuel exposure of 1300 Mwd/ton. These drivers and their targets are the components of a tube-and-rod coproducer assembly. The driver element is a coextruded Zircaloy-2 clad uranium tube with a 2.29 in. OD and 1.44 in. ID. The 0.565 in. diameter inner rod is the target element composed of Li-Al alloy, double canned in aluminum and Zircaloy.

Fuel swelling of the 1.25% enriched driver was determined by pre- and postirradiation bulk density measurements. The fuel (normal "N" composition, iron and silicon additions) in four drivers in the center of the charge swelled in excess of 2 vol%. A maximum swelling of 3.3 vol% was observed in an element in which the fuel is estimated to have operated with a maximum temperature of 530 C to an exposure of 1850 Mwd/ton. The estimated fuel exposure and the measured fuel swelling for the 13 driver elements is shown in Figure 2.17. The driver element showing the maximum swelling will be shipped to Radiometallurgy for detailed examination and evaluation.

Pre- and postirradiation density measurements made on the 13 target elements revealed no significant swelling of the target alloy. Radiometallurgical examination of the targets is in progress. The outer Zircaloy can from one of the target elements has been stripped without evidence of bonding to the inner aluminum can. Comparison of pre- and postirradiation density measurements made on the target alloy indicate no swelling.

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FIGURE 2.17

Fuel Exposure and Fuel Swelling
of Coproduct 1.25% Enriched Driver Elements

Identification of Irradiated Fuel Elements - E. A. Smith

Methods of distinguishing between natural and enriched aluminum clad fuel elements after irradiation are being explored. It is required to separate the two types at the pickup chutes in the reactor discharge basins. The present means of distinction, that of fabricating each type to a characteristic length, is undesirably expensive, so that a basis of separation other than length is needed.

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Preliminary tests were run to determine whether surface marks shallow enough to pass acceptance specifications could be seen and recognized under the lighting conditions normally obtained at the pickup chutes. Surface markings tried included coarse and fine knurls, circumferential grooves, spiral indentations, and sand-blasted surface. None of these could be distinguished at the bottom of the basin.

Inasmuch as Zircaloy-jacketed elements do not normally become encrusted with film during irradiation as aluminum jackets do, it seems possible that a band of zirconium plated on the aluminum jacket might show up after irradiation as a lighter or darker area than the remainder of the jacket. To test this possibility, three aluminum dummy slugs were fitted with a pressed-on thin band of Zircaloy at one end. These are being placed in 100-K flow loops (where film normally forms on aluminum jackets) to see if the Zircaloy bands have any effect on film deposition. If the results are favorable, methods of plating the aluminum with zirconium will be explored; if unfavorable, the use of ultrasonic or amplified microformer scanning of surface markings may be explored. These latter approaches can certainly be made to work; however, the time and expense of development will be greater than for the simpler visual methods.

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III. METALLIC FUELS DEVELOPMENT

Development work has continued on the program to develop the techniques required to fabricate coextruded, Zircaloy clad, and fully bonded Th-U alloy fuel elements for reactor service in pressurized, high temperature water environments. The irradiation behavior of these elements is being evaluated and additional elements and fuel alloys are being subjected to out-of-reactor destructive and nondestructive tests.

Additional development work has been included in this program for FY 64 on uranium base fuel materials having potential application for high temperature service.

Irradiation Testing of Thorium-Uranium Fuel Elements - R. K. Marshall

Irradiation testing of thorium-uranium fuel elements continued in the high-temperature, high-pressure, water-cooled ETR-P7 facility. Since the irradiation began in April 1963, the fuel elements have been irradiated for ten cycles and have attained an exposure of 0.9 at.% burnup. Goal exposure is 3 at.% burnup.

The three tubular test elements are Zircaloy-2 clad Th-2.35 wt% U^{235} -1 wt% Zr alloy fuel 4.45 cm (1.75 in.) OD x 2.67 cm (1.05 in.) ID. The test facility has operated at 204 C (400 F) coolant inlet at a pressure of 110 atm (1600 psi). Of the three 20 cm (8 in.) long elements, which are positioned vertically in the test, the central element, operating at the highest power, has accumulated a maximum of 2.7×10^{20} fissions/cm³ (8000 Mwd/ton). The maximum operating conditions and measured volume changes for the irradiation period are given in Table 3.1 and are plotted in Figure 3.1. The fuel swelling is low compared to equivalent test conditions for uranium. The 0.9% volume increase observed after Cycle 64 is slightly less than the theoretical minimum volume expansion due to fission product production.

TABLE 3.1

IRRADIATION HISTORY OF TUBULAR Th-2.5 wt% U-1.0 wt% Zr FUEL ELEMENTS

ETR Cycle	Burnup, fissions/cm ³ (Mwd/ton)	Temp. C	Maximum Operating Conditions		
			Surface Heat Flux, cal/(sec)(cm ²) (Btu/(hr)(ft ²))	Specific Power, w/g (kw/ft)	Fuel Volume Increase, %
54	2.8 x 10 ¹⁹ (800)	585	75 (1.0 x 10 ⁶)	69 (210)	0.15
55	5.4 x 10 ¹⁹ (1550)	538	68 (9.0 x 10 ⁵)	62 (188)	0.15
56	8.6 x 10 ¹⁹ (2475)	523	65 (8.6 x 10 ⁵)	59 (180)	Not weighed
57	1.2 x 10 ²⁰ (3400)	585	75 (1.0 x 10 ⁶)	69 (210)	0.5
58	Not irradiated				
59	1.2 x 10 ²⁰ (3500)	460	52 (6.9 x 10 ⁵)	46 (141)	0.5
60	1.6 x 10 ²⁰ (4500)	540	65 (8.7 x 10 ⁵)	60 (183)	0.5
61	1.8 x 10 ²⁰ (5200)	540	65 (8.7 x 10 ⁵)	60 (183)	0.8
62	2.1 x 10 ²⁰ (6200)	520	62 (8.3 x 10 ⁵)	57 (174)	0.8
63	2.5 x 10 ²⁰ (7400)	520	62 (8.3 x 10 ⁵)	57 (174)	0.8
64	2.7 x 10 ²⁰ (8000)	480	53 (7.1 x 10 ⁵)	49 (150)	0.8

FIGURE 3.1
Irradiation Behavior
of Th-2.5 wt% U-1.0 wt% Zr
Zircaloy-2 Clad
Fuel Elements

Th-U-Zr Alloys - R. S. Kemper

An additional coextrusion of a Th-2.5 wt% U-1 wt% Zr alloy billet in Zircaloy-2 was made to provide rod stock for fabrication tests to determine the feasibility of applying the hot headed and projection welded closure to Zircaloy clad thorium base fuels. Coextrusion was done with 750 C, 2 hr preheat of a $2\frac{1}{4}$ in. diameter billet using a 0.623 in. diameter die for a reduction of 13.4 to 1. Extrusion constants were 18.9 tsi maximum and 17.9 tsi running. This is a lower reduction ratio than has been used previously and the degree of clad-core bonding is of interest. Examination of as-extruded specimens is in progress.

Sections of ten coextruded Th-U-Zr alloys containing up to 5 wt% U and 10 wt% Zr were heat treated at 800 and 900 C to determine conditions required for heat treatment of samples for defect corrosion testing.

Th-U-Zr alloys containing either beryllium or aluminum additions were all fabricated to sheet stock approximately 0.050 in. thick by hot rolling in stainless steel compacts. Specimens have been prepared and sealed in evacuated capsules for heat treating.

Irradiation Testing of Metallic Uranium with Submicron Uranium Carbide Dispersions - J. W. Weber and R. K. Marshall

The purpose of the irradiation test is to investigate the ability of submicron dispersion of uranium carbide to reduce uranium swelling. Three NaK-filled capsules will be irradiated in the ETR to determine the irradiation performance of fuel rods produced from chill-cast uranium shot relative to rods produced from slowly cooled arc-melted uranium. Of these three capsules, two will compare the swelling performance of two fuel rods identical in uranium composition, but with one having a uranium carbide size of 2 to 5 μ produced from arc-melted uranium and the other a carbide size of less than 0.5 μ produced from the uranium shot. The third capsule contains two fuel rods with the fine carbide.

One capsule which contains fine and coarse carbide fuel rods has attained goal exposure of 0.3 at.% burnup. The capsule operated at a maximum temperature of 550 C.

The other two capsules, one of which contains fine carbide dispersion specimens only and one which contains both fine and coarse dispersion specimens, were recently charged into the ETR. Both capsules are operating within the test design range of 600 to 625 C.

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