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REPORT TO THE WORKING COMMITTEE

FROM HANFORD

by:

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April 12, 1965

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CURRENT REACTOR PRODUCTION FUELS

Uranium Core Production Data

Quality of Uranium Cores

Inspection of Virgin Cores

The combined bare core reject rate (visual and UT-2) was 0.19 per cent for the period December 1, 1964 through March 31, 1965.

Grain Size

The average grain size of cores received during the period December 1, 1964 through January 31, 1965 was between A-6 and A-7 for both natural and enriched uranium.

Uranium Specifications

New product specifications for I&E cores have been issued. These are a general updating of previous specifications, and the major changes are in establishing minimum levels for iron and silicon of 100 ppm and 75 ppm, respectively.

The use of a new casting cycle at National Lead Company of Ohio for use on I&E ingots has been approved. The new cycle changes the mold heating with subsequent improvement in the soundness of ingots. No differences could be found in the distribution of impurities in ingots cast under the old and the new mold heating cycle. The change consists of eliminating the mold heating power during the B and C phases of the casting cycle.

Fuel Performance

Twenty-one (21) normal and five (5) enriched production fuel elements failed during the period December 1, 1964 through March 31, 1965. In addition, five (5) normal overbore elements failed in C Reactor. A tabulation of the failures is presented in Table I.

The overbore failures have all resulted from poor end bonding, a problem which was reported in the Working Committee Report of April, 1964. Hot die sized overbore elements are being phased in to replace the ALSi overbore. The end bond strength and integrity of hot die sized elements is greater than ALSi elements. The initial charging of hot die sized overbore elements will be in April, 1965.

The warp and bond count trends are projected in Figures 1 and 2. The bond count curves have been plotted separately for the "O" models and the "C" and "K" models. This was done as "K" models show a consistently higher bond count than the "O" models. Both curves show a slight downward trend in bond count since the last reporting period.

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PRODUCTION RUPTURES
I&E FUEL ELEMENTS

<u>Fuel Element Type</u>	<u>Reactor</u>	<u>Per Cent of Goal Exposure</u>	<u>Rupture Classification</u>	<u>Rupture Date</u>
Enriched Self Support	KE	120	Side Ledge Corrosion	12/4/64
Enriched Nonbumped	F	89	Side Hot Spot	12/9/64
Normal Nonbumped	B	128	Not Examined	12/13/64
Normal Nonbumped	B	46	Not Examined	12/6/64
Normal Nonbumped	DR	23	Not Examined	12/21/64
Normal Nonbumped	DR	22	Side Hot Spot	12/23/64
Normal Nonbumped	DR	59	Not Examined	12/25/64
Normal Nonbumped	C	39	Side Hot Spot	12/31/64
Normal Nonbumped	D	40	Side Hot Spot	1/2/65
Normal Nonbumped	H	42	Side Hot Spot	1/6/65
Normal Nonbumped	B	95	Side Hot Spot	1/7/65
Normal Nonbumped	D	48	Side Hot Spot	1/13/65
Normal Overbore	C	78	End Manufactur- ing Defect	1/13/65
Normal Nonbumped	D	60	Side Unclassified	1/15/65
Normal Overbore	C	77	End Manufactur- ing Defect	1/16/65
Normal Overbore	C	57	End Manufactur- ing Defect	1/17/65
Normal Nonbumped	D	58	Side Hot Spot	1/19/65
Normal Nonbumped	H	64	Side Hot Spot	1/19/65
Normal Nonbumped	B	96	Side Hot Spot	1/31/65
Normal Nonbumped	H	89	Not Examined	2/6/65

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TABLE I (Cont'd)

<u>Fuel Element Type</u>	<u>Reactor</u>	<u>Per Cent of Goal Exposure</u>	<u>Rupture Classification</u>	<u>Failure Date</u>
Normal Nonbumped	D	95	Side Hot Spot	2/10/65
Normal Self Support	KE	27	End Manufactur- ing Defect	2/13/65
Normal Overbore	C	11	End Manufactur- ing Defect	2/15/65
Normal Nonbumped	D	95	Side Unclassified	2/18/65
Normal Nonbumped	F	86	Side Hot Spot	2/22/65
Normal Nonbumped	H	21	Side Hot Spot	2/23/65
Enriched Nonbumped	H	31	Side Hot Spot	3/13/65
Normal Nonbumped	H	34	Side Hot Spot	3/4/65
Enriched Bumped	B	94	Side Unclassified	3/11/65
Normal Overbore	C	52	End Manufactur- ing Defect	3/14/65
Enriched Nonbumped	B	75	Side Hot Spot	3/25/65

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FIGURE 1
CANNING DATE vs $\sqrt{WARP}$

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AVERAGE WARP ($\sqrt{\text{Mils}}$)

1964

CANNING DATE

1963

1962

UNTRANSFORMED AVERAGE

RL-REA-1005

FIGURE 2

CANNING DATE vs. BOND COUNT

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AVERAGE BOND COUNT-LOG (X+1)

Induction Heat Treating

A partially untransformed induction heat treated O3N core was detected by the UT-2 testers in January. Metallographic examination showed a fine grained area (finer than A-9) extending part way around the unnumbered end of the core. The rest of the core had a normal A-6 grain size.

Three samples were taken from the core for an X-ray study. A half-inch section was taken from the abnormal end, and from this two small pieces representing the small grain size and the normal grain size were cut. A half inch section was cut from the normal end, and X-ray scans were made on the three samples. The results are shown in Table II. An analysis of the X-ray data shows there is texture, similar to that found in untransformed uranium, in both of the pieces from the "abnormal end", but it is more pronounced in the small-grained section. The data from the numbered "normal end" is typical of a normal beta heat treated core.

Thoria Program

As of April 6, 1965, 123 tons of thoria have been charged into the Hanford reactors and 36 tons have been discharged. Eleven thoria elements have failed during the program, and all of the failures examined to date are associated with water entry through the weld. On nine failures examined, two distinctly different types of deformation of the cladding have been observed. Three of the failures had a bulge on the cap end in the area annealed during welding, and the bulge appears to have been caused by internal gas or steam pressure. The other six failures are characterized by more or less uniform swelling along the length of the element, and the cladding on some was ruptured longitudinally. The uniform swelling appears to be caused by expansion of the thoria when water comes in contact it under reactor conditions, and in these elements the thoria has become hard with the general appearance of concrete. Table III summarizes the pertinent information on thoria failures.

In an effort to duplicate the uniform swelling with out-of-reactor tests, elements with defective welds were autoclaved under two sets of conditions:

	<u>Temperature</u>	<u>Pressure</u>	<u>Time</u>
Condition 1	175 C	300 psi	24 hours
Condition 2	70 C	15 psi	72 hours

The elements tested under this condition showed an OD increase of 0.149 to 0.178 inches on the base end, and an OD increase of 0.010 to 0.029 inches on the cap end. The element tested under Condition 2 showed an OD growth of 0.010 inches on the base end and no change on the cap end. In all cases, the growth appeared to be caused by a buildup of a layer of corrosion product up to 0.11 inches thick on the inside of the can. The corrosion product layer was identified by X-ray diffraction as $\text{Al}_2\text{O}_3(\text{OH})_2$ with ThO_2 included. Similar corrosion has not been observed in any of the reactor failures examined to date. Although the mechanism of the autoclave swelling is different than the reactor swelling, it was felt that both may have a common controlling

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TABLE II

X-RAY RESULTS ON
PARTIALLY UNTRANSFORMED
INDUCTION HEAT TREATED CORE

Longitudinal Texture Coefficients

<u>Plane hkl</u>	<u>Numbered End (Transformed)</u>	<u>Unnumbered End (Untransformed)</u>	
		<u>Fine Grain (> A-9)</u>	<u>Normal Grain (A-6)</u>
020	1.14	2.83	1.86
110	.94	2.42	1.88
021	1.07	0.72	1.34
002	1.67	0.24	1.50
111	0.85	1.01	0.95
112	0.73	0.04	0.33
130	1.34	1.51	2.28
131	0.87	1.89	0.90
023	1.00	0.24	0.87
200	0.62	2.28	1.12
113	0.93	0.15	0.56
132	1.22	0.50	0.47
133	1.01	0.15	0.28
114	0.94	0.17	0.47
150	0.99	1.53	1.55
223	1.16	0.31	0.72
152	0.78	1.04	1.14
312	1.15	2.03	1.21
Longitudinal G2	0.016	-0.051	0.023

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TABLE III
THORIA FAILURES

Failure Date	Residence Time (Days)	Reactor	Hanford Lot No.	Element Density (g/cc)	Failure Type (1)	Oxide Lot No. (2)	Oxide Analyses (Vendor-Supplied Sample)		
							Vendor LOI (%)	Hanford LOI (%)	Hanford Offgas (cc/g) (3)
1/19/65	51	KW	TK-1808	6.8	B	T-193 T-194	0.05 0.11	0.03 0.03	0.154 0.431
1/19/65	51	KW	TK-1010	7.1	U	T-115	0.08	0.06	
2/22/65	30	KW	TK-4340	7.4	U	T-265 T-267	0.06 0.06	0.03 0.03	
2/22/65	30	KW			U				
2/22/65	49	KW			N				
2/18/65	49	KE	TC-2250	7.5	B	N-6	0.03	0.04	0.291
3/1/65	60	KE	TC-2650	7.5	N	N-9	0.01	0.06	0.210
3/25/65	91	C	TC-1650	7.5	U	N-7 T-209	0.01 0.04	0.05 0.03	0.133
3/29/65	65	KW	TK-4440	7.4	U	T-263 T-272	0.06 0.10	0.03 0.07	
3/29/65	65	KW			B				
4/1/65	91	KE			N				

(1) Code for failure types: B - A buldge at the cap end
U - Uniform swelling along element length
N - Not examined

(2) Lot Numbers : Mallinckrodt Chemical Works lots have letter "T"
Nuclear Fuels Services, Inc., lots have letter "N"

(3) Offgas Analyses : Vacuum extraction at 1200 C

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factor which could be a guide to the "reactivity" of the oxide. The NFS firing temperature of 1060 C is considered to be borderline for producing a dense, non-reactive oxide. Therefore, a sample of NFS lot 42 was fired at 1100 C for four hours, and an element was tested under autoclave condition 1 along with an element from the as received oxide in lot 42. The oxide analyses before and after firing and the autoclave test results are tabulated below:

Autoclave Tests on NFS Lot 42

	OD Growth			BET Surface Area m ² /g*	Offgas Volume cc/g
	Cap	Middle	Base		
As Received	0.027"	0.036"	0.060"	2.26	0.301
After 1100 C-4 hours	0.004"	0.021"	0.215"	0.124	0.022

* Average of two analyses

As with previous tests, the OD growth was a result of corrosion of the inside of the can, and the additional firing did not reduce the amount of corrosion, even though the surface area was reduced by a factor of twenty. The sodium content of the lots involved in the autoclave tests is from 200 to 385 ppm.

As the failures are associated with water entry through the weld, action has been taken to improve the weld testing and a program to develop a method for providing double closures has been started. Since March 22, the welds have been tested by helium leak checking in addition to the bubble testing and visual inspection which have been in use since the start of the thoria program.

N-RD PRODUCTION FUELS

N-Reactor Fuels: Production Status

During the first two months of CY 1965, 122 tons of billets were coextruded with a uranium utilization of 75 per cent for January and 63 per cent for February. The below-normal figure for February is the result of the large amounts of scrap generated from reworking 460 outer tubes from 24 inches to 12 inches length and losses from restraightening warped inner tubes. The material involved in both programs has been shown in an "in-process" category and heretofore did not reflect on over-all uranium utilization values until the reprocessing was affected.

No unusual fabrication problems were noted during the quarter. The chronic problem of inner fuel warp after heat treat persists. Work has been directed toward a second restraightening process and heat treat. The yield on the first 147 pieces restraightened was 56 per cent. Effort to further improve recovery of warped inner pieces is continuing.

Coextrusion of Eccentric Billets

In the fabrication of uranium billets for NPR coextrusions, some billets are rejected for excessive wall thickness variation. A study was initiated to

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determine the effect of coextruding billets with wall thickness variation between the current specification limit of 0.030 inch maximum and an arbitrary test limit of 0.040 inch. Controls for the 20 test NIE billets were 20 standard production billets with wall thickness variation less than 0.011 inch.

The forty billets were randomly mixed and processed through the Shop. The billet assembly, the billet thickwall marks were given various orientations relative to the evacuation tube which led to known thickwall orientation in the press liner. The coextruded fuel pieces were measured for warp and wall thickness variation. The truck cards were compared for all categories of reject. The study showed no significant differences in reject rates nor any appreciable difference in warp after heat treatment. Large differences were seen, however, in wall thickness variation between the two groups. For comparison, two groups of fuel pieces are shown in the following table. The first group is endmost fuels (1, 2, 17, 18) and the second group is middle pieces (7, 8, 9, 10, 11). The groups are compared for average wall thickness variation and standard deviation for position in the coextrusion and for two press orientations of the billet thickwall. The level of statistical significance is given for the difference.

TABLE IV

The Effect of Billet Wall Thickness
Variation on Wall Thickness Variation of Coextruded Fuel Elements

Billet Lot	Wall Thickness Variation in Coextruded Fuel Elements (mils)			
	Fuel Pieces 1, 2, 17, 18		Fuel Pieces 7, 8, 9, 10, 11	
	<u>X</u>	<u>σ</u>	<u>X</u>	<u>σ</u>
Test (5)	9.5	3.9 ⁽¹⁾	10.8	3.3 ⁽²⁾
Control (6)	12.3	8.5 ⁽¹⁾	4.4	2.3 ⁽²⁾
0° Test (7)	6.6	3.0 ⁽³⁾	9.45	2.14 ⁽⁴⁾
180° Test (8)	11.0	3.9 ⁽³⁾	9.35	1.72 ⁽⁴⁾

- Notes:
- 1) Significant difference at 78% level.
 - 2) Significant difference at 99.99% level.
 - 3) Significant difference at 99.99% level.
 - 4) No significant difference.
 - 5) Wall thickness variation in billets was 0.030 to 0.040 inch.
 - 6) Wall thickness variation in billets was less than 0.011 inch.
 - 7) The thickwall of test and control billets was at the top of the liner.
 - 8) The thickwall of test and control billets was at the bottom of the liner.

The data summarized in the above table show that a Class II specification for wall thickness variation is feasible and such a specification is now in effect. The data also show that billets of higher wall thickness variation should be oriented in the coextrusion press liner with the thickwall at the top of the liner.

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Improved Uranium Utilization

One method of improving uranium utilization from coextrusion assemblies is to increase the total length of the coextruded fuel tube without a proportionate increase in the length of the end defects. One method of obtaining the increased coextruded length is to increase the diameter of the billet and another is to increase the length of the billet. Test coextrusions have been made that demonstrate both increases are feasible, and could lead to as much as a 60 per cent increase in the length of the coextruded fuel tube.

Four N-outer and four simulated driver tube coextrusions were made from an 8.1 inch diameter coextrusion liner instead of the standard 7-1/2 inch liner. The upset and run pressures for the large diameter coextrusions were below 2,700 tons and the extruded surfaces appeared excellent. Evaluation of metallurgical and dimensional properties is underway. Use of this billet design should increase the design utilization of the uranium from 86.6 to 87.5 per cent.

A "two-step" coextrusion process that enables the use of longer billets has been demonstrated. Using this process, the maximum length of the billet can be increased by not inserting a graphite follower block during the first step of the coextrusion. The coextrusion is stopped while the die cone is still filled with materials from the rear of the billet assembly. At this point, the mandrel is retracted and a graphite cutoff is inserted behind the die cone and the coextrusion completed in the second step. The process has been designed to assure that all material that will subsequently be contained within the accepted fuel elements has cleared the die during the first step of the cycle.

Fourteen "two-step" N-outer coextrusions were completed during the month and are being processed through the balance of the finishing operations. The evaluations completed to date indicate that the two-step process can be adopted for routine processing. Warp on the two-step coextrusions averaged 10.6 mils while the control extrusion warp averaged 10.4 mils. Delay time between the first and second cycles averaged 2.5 minutes and the extrusion pressure for the second step averaged 250 tons below the upset pressure for the first step. Larger demonstration tests of both the two-step and large diameter billet processes are planned.

Using this process, billet length can be increased three inches and this will increase uranium utilization from 86.6 to 88.6 per cent.

Fabrication of Uranium Billets

Development work on uranium fabrication problems has led to two alternate fabrication routes which give promise of significantly reducing offsite fabrication costs. Flow sheets comparing the present (standard) process with the alternates (direct cast and upset forge) are given in Table V. Both alternate routes will result in reduced uranium fabrication costs. At the present time, accurate cost data are not available on the alternate routes. However, the potential cost reductions can be summarized as follows:

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TABLE V

ALTERNATE BILLET FABRICATION ROUTES

Standard

Uranium Utilization:

NOE - 61.8%
NIE - 57.8%

Direct Cast

Uranium Utilization:

Cast and Machine:

NOE - 67.6%
NIE - 74.0%

Cast to Size:

NOE - 70.8%
NIE - 79.4%

Advantages:

1. Eliminate the primary extrusion process
2. More amenable to higher alloy contents in the uranium.

Disadvantages:

1. Low reduction ratios - less breakup of non-metallic inclusions
2. No long-term coextrusion Experience

Upset Forge

Uranium Utilization:

NOE - 79.2%
NIE - 84.6%

Advantages:

1. Highest metal utilization
2. Good economics
3. Good coextrusion experience in wrought metal route
4. Chip-free process

Disadvantages:

1. Need primary extrusion press
2. Difficult inspection of upset surface
3. No experience with the problem of bonding unmachined uranium surfaces to the zircaloy cladding.

<u>Standard Operation</u>	<u>Direct Cast</u>	<u>Upset-Forging</u>
Ingot Casting	More casting operations will be required	Since elimination of chips will allow heavier ingots to be cast, less castings will be required
Primary extrusion	Eliminated completely	Less operations
Extrusion cutup	Same as standard	Same as standard
Billet machining	Same as standard	Eliminated completely
Upset forge		New operation
Chip briquetting	Same as standard	Practically eliminated

Preliminary estimates show that a potential cost reduction in the order of \$500,000 to \$1,000,000 annually may be realized.

A major revision to the fabrication process creates concern about potential effects on metal quality. Since both alternates propose revision to the standard process, evaluation of resultant uranium metal quality will be required.

Alternate Coextrusion Facility

In the event of a major press damage at HAPO, arrangements are being made to have a "backup" coextrusion facility available at Reactive Metals in Ashtabula, Ohio. During any prolonged press outage at HAPO, billets would be assembled and shipped to Ashtabula for coextrusion and returned to HAPO for finishing. Preliminary estimates are that approximately \$13,000 will provide all the backup tooling necessary at Ashtabula and will also finance the cost of demonstrating the process on about 20 coextrusions. Plans are to complete this demonstration run and to have the backup tooling in place at Ashtabula by July, 1965.

Preliminary tool design specifications and operating procedures were discussed in detail with RMI personnel during February. No major problems were encountered and work is proceeding on schedule.

REACTOR FUEL DEVELOPMENT

N-Reactor Fuels Development

Process Development

Alternate Uranium Compositions

Both the operating conditions for driver fuel elements of a single tube design and the extended goal exposure of the current tube-in-tube design require development of a fuel material that is more swelling resistant than the current 150 ppm Fe + 100 ppm Si composition. Driver elements of the single-tube design having the compositions given in Table VI were

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beta and gamma heat treated by the salt-salt process and the following evaluation tests are in progress: 1) examination of microstructure and bond zone, 2) hot hardness tests at 300 and 400 C, 3) determination of recrystallization temperature after 21.4 per cent reduction in area, and 4) stress cycling tests at 400 and 500 C. These tests will be completed in April, 1965.

TABLE VI

Fuel Alloys Coextruded to the
Single Tube Driver Geometry

<u>Extrusion Number</u>	<u>Billet Number</u>	<u>Fe (ppm)</u>	<u>Si (ppm)</u>	<u>Al (ppm)</u>	<u>Alloy Designation (No.)</u>
X-161	37785C	150	100	---	1
X-164	25018A	250	350	---	2
X-167	25016C	250	250	250	3
X-162	25014C	350	---	800	4
X-192	20584A	350	350	800	5

Hot hardness data are given in Table VII, and stress-cycling tests completed to date are shown in Table VIII. It is planned to run the stress-cycling tests to 5000 cycles for each material and examine the microstructure for intergranular cavities and tears.

As one basis for comparison of these five alloys, the number of intermetallic compounds of size 0.02 microns and the compositions of each compound species was calculated; these results are shown in Table IX. Several simplifying assumptions were made; the two most limiting assumptions were that all compound particles were of equal size and that all had been precipitated from solution. If pinning of dislocations by second phase particles is primarily responsible for both elevated temperature deformation behavior and irradiation performance, the five alloys fall in three groups distinguished by the number of particles in each: Alloy 1; Alloys 2, 3, and 4; and Alloy 5. In support of this generalization, the hot hardness values that are shown in Table VII are similar for alloys 2, 3, and 4 and are different than those hot hardness values reported for the standard alloy. Slight increases are generally observed for gamma treated material in which all the alloy addition is taken into solution at heat treating temperature. The stress cycling data that are now available to date show only that those alloys containing aluminum have greater resistance to plastic deformation at 400 and 500 C.

Large-scale fabrication tests of the same uranium alloys for subsequent irradiation testing in N-Reactor have continued. Coextrusion tests to date have shown that all the alloys result in as good or better clad thickness variation control than obtained with standard production alloy provided some beta heat treatment of the uranium in either ingot or billet form has been performed. Use of uranium that had not been heat treated resulted in severe wrinkling of the cladding. Coextrusion of

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TABLE VII

HOT HARDNESS OF URANIUM FUEL MATERIALS

Material	Alloy Designation (No.)	Heat Treatment	Hardness* (Kg/mm ²)	
			300 C	400 C
U+150 ppm Fe+100 ppm Si	1	Beta	94.6	60.3
		Gamma	93.1	60.8
U+250 ppm Fe+350 ppm Si	2	Beta	98.5	70.8
		Gamma	101.7	75.1
U+250 ppm Fe+250 ppm Si+250 ppm Al	3	Beta	94.8	71.7
		Gamma	96.9	75.0
U+350 ppm Fe+800 ppm Al	4	Beta	105.0	68.8
		Gamma	101.2	74.6

* Average of tests run at 400, 600, and 800 pound loads with a four hour loading time and a 10 mm diameter WC ball.

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TABLE VIII

ELEVATED TEMPERATURE STRESS CYCLING TESTS OF URANIUM FUEL MATERIAL

Material	Alloy Designation (No.)	Heat Treatment	No. of Cycles	Total Strain at Temperature (Per Cent in 1.125 Inch Gage Length*)	
				400 C**	500 C***
U+150 ppm Fe+100 ppm Si	1	Beta	500	.71	.53
		"	1,000	.98	1.69
		"	5,000	4.62	--
		Gamma	500	.44	1.51
		"	1,000	.62	1.51
U+250 ppm Fe+350 ppm Si	2	Beta	500	.27	.89
		"	1,000	1.15	2.66
		Gamma	500	.35	.27
		"	1,000	2.04	.62
U+250 ppm Fe+250 ppm Si +250 ppm Al	3	Beta	500	.27	.09
		"	1,000	.44	.27
		Gamma	500	.09	.09
		"	1,000	.18	.35
U+350 ppm Fe+800 ppm Al	4	Beta	500	0	.35
		"	1,000	.18	.35
		Gamma	500	.27	.09
		"	1,000	.35	.62

* Deformation measured on specimen after cooling to room temperature.

** Stress level 30,000 psi - cyclic stress $\pm$ 3000 psi.

*** Stress level 15,000 psi - cyclic stress $\pm$ 3000 psi.

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TABLE IX

CALCULATED NUMBER OF INTERMETALLIC COMPOUND PARTICLES IN URANIUM BASE ALLOYS

Alloy Designation (No.)	Uranium Composition			Number of Cubic-Shaped Particles With Edge Dimensions of 0.2 Micron per cc of Fuel ($\times 10^{12}$)				Volume Per Cent Compound	Total Number of Particles per cc ($\times 10^{12}$)*
	Fe ppm	Si ppm	Al ppm	$U_3(Si)$	$U_6Fe(Si)$	$UAl_2(Fe)$	Total		
1	150	100	---	0.098	0.674	---	0.772	0.62	0.098
2	250	350	---	1.11	1.15	---	2.26	1.81	1.11
3	250	250	250	0.69	1.09	0.33	2.11	1.69	0.99
4	350	---	800	---	0.89	1.29	2.18	1.74	1.29
5	350	350	800	1.08	1.40	1.25	3.73	2.99	2.33

* Assumes $U_6Fe(Si)$ grows rapidly to a size that is not effective in pinning either gas bubbles or dislocations and is also not effective in strengthening or hardening.

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these alloys for the single driver element has been done from a 7-1/2 inch diameter coextrusion liner with a tube fuel reduction ratio of 14.1/1. The as-coextruded bonds were weak because of the low reduction ratio. Further coextrusion will be done from an 8.1 inch diameter liner to obtain a larger reduction ratio, 16.2/1.

Uranium fabrication routes tested for the alloys included casting to 13, 11, and 7-1/2 inch diameters. Primary extrusion of 13 and 11 inch diameters has been tested, both before and after beta heat treatment. To date, single beta heat treatment and water quench in either the ingot or the billet form have been successful for both the standard alloy and for the 350 ppm Fe + 800 ppm Al alloy. Some cracking problems have been encountered in alloys containing high amounts of silicon, particularly when heat treated as 13 inch and 11 inch castings and water quenched; heat treatment of these alloys is continuing at the Feed Materials Production Center.

Thermal Expansion of N-Reactor Fuels

To obtain information necessary for the design of supports for N-fuel components, measurements of the diametral thermal expansion of beta heat treated N outer fuel tubes (NOE's) and single-tube driver elements have been made. In the temperature range 20 to 300 C, the diametral thermal expansion coefficient for both NOE's and single tubes is 15.5 ± 0.5 in/in/°C. This compares with previously determined coefficients of longitudinal thermal expansion for NOE's and N inner fuel tubes (NIE's) of about 13.0 in/in/°C (range 25 to 100 C). To determine the extent to which the Zircaloy-2 clad influences the coefficient of the fuel-clad composite, the diametral thermal expansion of the uranium core of an NOE fuel tube was also determined after machine removal of the inner and outer Zircaloy-2 clad. In the temperature range 20 to 300 C, the coefficient of the declad fuel ranged from 18.0 to 19.5×10^{-6} in/in/°C which is, on the average, about 20 per cent greater than that for the clad-fuel composite.

Target Element Development

Zircaloy-2 Corrosion in Molten Aluminum

The corrosion of Zircaloy-2 in molten aluminum is being studied in the temperature range 700 C to 1100 C. Corrosion of the Zircaloy is nearly uniform and the approximate penetration rates at 700 C and 900 C are 0.4 and 3.5 mils per minute, respectively.

Experiments to date show that the corrosion rate slows somewhat with time. This results from the buildup of zirconium:aluminum intermetallics in the molten bath and, more importantly, on the surface of the zircaloy coupons ($ZrAl_3$). The presence of an initial autoclave film also slows the corrosion rates slightly. The following data illustrate the previous statements:

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<u>Temperature (°C)</u>	<u>Time (Minutes)</u>	<u>Average Penetration Rate (Mils/Min.)</u>	
		<u>Autoclaved Surface</u>	<u>Etched Surface</u>
700	5	0.7	1.0
"	30	0.18	0.42
"	50	0.35	0.45
"	80	0.25	0.27
"	100	0.31	--
900	5	4.6	3.1
"	9	3.0	3.2

Experiments are conducted in clay crucibles using resistance heating at 700 and 800 C and induction heating from 900 to 1100 C. Air atmosphere covers the molten aluminum.

Protective Coatings for Zircaloy-2

Flame-sprayed coatings, nickel plating, nickel, and silicon additions to the molten aluminum are being evaluated as deterrents to the corrosion of Zircaloy-2 by molten aluminum. Nickel and silicon additions of 3 and 12 wt.%, respectively, affect a distinct reduction in corrosion rate. Flame-sprayed coatings that are protective in pure aluminum at 700 C are tungsten carbide, zirconia, molybdenum, tantalum, and most notably, alumina. Coatings that do not perform well are nickel plate, steel, nickel, aluminum-iron bronze, aluminum, and copper. The coatings which performed well at 700 C will be re-evaluated at 1100 C.

Irradiation Experience

Uranium Alloy Capsule Irradiations

Studies are in progress to evaluate both the in-reactor performance and the subsequent irradiated corrosion behavior of deliberately defected samples as a function alloy composition and pre-irradiation treatment. Forty test capsules and five temperature monitoring capsules were charged in five process tubes in a production reactor. The irradiation includes 120 test specimens of 17 uranium-base alloys containing small additions of Zr, Nb, Mo, Al, Si, Fe, P, and C in various heat treated conditions. The compositions and heat treatments have been described in previous Hanford FEDC contributions. All thermocouple systems are operating satisfactorily and the initial data show maximum uranium temperatures at full reactor power are in the range of 470 to 490 C. These temperatures are below the desired 625 C and efforts will be made to raise them by changes in the surrounding reactor fuel loading and water flow in the process tubes containing the capsules. A goal exposure of 3000 to 4000 MWD/T is planned for these capsule fuel core materials.

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Post-KER Irradiation Evaluations of N-Fuels

Correlation of Post-Irradiation Nondestructive Test Indications

Indications of defects in irradiated N-Reactor fuel components have been obtained using ultrasonic techniques. Radiometallurgical examinations have been made upon selected components with several kinds of indicated defects and the existence of the defects has been verified. However, the defects are believed to be caused by post-irradiation handling and not induced by irradiation. In one case, ultrasonic nondestructive testing indicated a defective cap or closure. The conclusion of the detailed destructive examination was that the closure was damaged during post-irradiation handling and that there was no pre-irradiation or irradiation-induced material or fabrication defect in the fuel element closure. In several other cases, in which large (two to three inches long) longitudinal cracks in the fuel in the end closure region were indicated by nondestructive tests, cracks were again disclosed by the metallographic examinations. Although these cracks are believed to be the result of post-irradiation handling prior to the ultrasonic testing, attempts to produce such cracks in uncracked irradiated elements by intentional rough handling before the ultrasonic tests were not successful.

Post-Irradiation Annealing Studies

Post-irradiation annealing at temperatures up to 700 C have been completed in the two different fuel compositions that were irradiated to 2700 MWD/T in the prototype geometry (K5E-5) of the single-driver element. The irradiation test and the post-irradiation annealing results both indicate that the fuel with iron and aluminum additions is much more resistant to swelling than is the fuel with iron and silicon additions. A tabulated summary of the irradiation and post-irradiation annealing behavior of the two fuel materials is shown below.

Fuel Composition	Volume Increase (Per Cent)			
	As-Irradiated 2700 MWD/T	Annealed		Total
		100 Hr. @ 625 C	100 Hr. @ 700 C	
U + 140 ppm Fe + 120 ppm Si	2.2	1.1	1.7	4.9
U + 390 ppm Fe + 840 ppm Al	1.2	0.7	-0.3	1.6

Post-irradiation metallography showed both grain-boundary tearing and crystallographically-aligned micro-porosity in the Fe and Si alloy, whereas the Fe and Al alloy showed only slight grain-boundary tearing and no aligned porosity. After the 625 C anneal, a coalescence of both types of porosity was evident in the Fe plus Si alloy, but little change was noted in the Fe plus Al alloy. After the second anneal (700 C), spherical porosity was observed in both the grains and grain boundaries of the Fe plus Si alloy. This additional porosity, which

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resulted in a further volume increase, is believed to result from coalescence of fission gases. After the second anneal of the Fe plus Al alloy, there was a small amount of porosity at grain boundaries, but this was probably just a coalescence of previous porosity as it did not produce a further volume increase. Both the metallographic observations and the density data indicate that fission gas mobility is inhibited to a greater extent in the Fe plus Al alloy than in the Fe plus Si alloy.

Radiometallurgical Examination of N-Fuel Supports

Both inner (buggy-spring type) and outer (suitcase-handle type) N-fuel component supports have been examined after stripping from components irradiated to tube exposures of 2000 MWD/T. Metallographic examinations and hydrogen analyses by hot vacuum extraction indicates the supports to be in good condition with the handle-type containing about 25 ppm hydrogen and the spring-type containing about 90 ppm hydrogen. The outer supports revealed an absence of hydride platelets or needles in the structure. The inner supports revealed a rather uniform distribution of hydrides somewhat oriented in the direction of rolling of the Zircaloy-2 support stock and a few platelets oriented perpendicular to the surface at the inside radius of sharp bends. The inner supports, however, do not represent those currently used as they had not received either the solution treatment or annealing process applied to more recent production for the purpose of controlling the orientation of precipitated hydride platelets.

Current Reactor Fuel Development

Alternate Process Development

The hot die sizing (HDS) process is being developed to replace the existing AlSi dip brazing process for the small and K Hanford reactors. Aluminum cladding is diffusion bonded to electronickel-plated uranium by the application of heat and pressure. The cylindrical surfaces of preheated fuel assemblies are bonded by pressure applied during sizing with cold dies. End bonding is accomplished in a following heating and pressing step.

Production of hot die size fuel in the pilot plant was started in January, 1965 with 5,250 "0"-size and 1,550 overbore fuel produced during the first quarter. Prototypic production equipment, including the nickel plating machine, preheat furnace, sizing press, and end bonding press are in operation. The nickel plating machine has exceeded design capacity and the new high-capacity preheat furnace and sizing press were put into service during the month of March.

Initial hot die size fuel elements displayed an unforeseen growth behavior during irradiation. Several irradiation tests have been conducted to determine the cause of the growth during irradiation. Results of the most recent irradiation test show that oil-quenched cores have greater dimensional stability than water-quenched cores. Hot die size fuel fabricated with oil-quenched cores had an average growth of 0.0034 inch compared to 0.0121 inch

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for matched water-quenched material. The average internal diameter change for oil-quenched and water-quenched fuel was 0.0001 inch and 0.0133 inch, respectively. Stress relieving the oil and water-quenched cores by alpha annealing and AlSi pretreatment either did not produce any significant changes or the changes were only slight and would not contribute to improved dimensional stability during irradiation. As a result of these findings, the heat-treating specifications for HDS cores have been changed to oil-quenching. Plans are underway for the testing of cores heat treated by the solid-oil, hollow-water, and induction-oil heat treating methods to achieve the best economic heat treating method for hot die size fuel cores.

Two other production tests were irradiated to determine the effect of the following variables upon hot die size fuel dimensional stability: (1) resistance-heated versus induction-heated end bonding, (2) end bonding pressure, (3) core dimension effect upon heat treatment. Test data showed that induction heated end bonding was the only variable which resulted in a significant but slight improvement in dimensional behavior. All HDS production tests have been irradiated to a goal exposure of about 800 MWD/T. Table X summarizes dimensional changes of hot die size fuel during irradiation testing. Internal and external diameter changes at the midpoint of hot die size fuel elements for the various production tests are illustrated in Figures 3 and 4.

Process development activities have been directed towards making the hot die sizing pilot plant operational. Primary effort was in the area of debugging the nickel plate machine and installing and debugging the increased capacity preheat furnace and sizing press. Limited testing was conducted and will resume when pilot plant equipment is fully operational. High speed sizing (approximately 280 inches per minute) and an improved closure technique have been demonstrated. The maximum holding period between core nickel plating and sizing has increased from 24 hours in a vacuum to 120 hours in a non-corrosive environment. There were no differences in bond strength for material held from 24 to 120 hours either in vacuum or in air.

A flow charge of integral ribbed fuel produced by the hot die sizing process has been fabricated. Self-support ribs were produced by sizing fuel through a slotted die.

Corrosion testing of high-silicon aluminum alloy (six to ten per cent Si) for hot die size fuel elements is almost complete. Evaluation will be completed after test coupons are removed from the reactor corrosion test loop. Preliminary corrosion data indicate that high-silicon aluminum alloy has as good or better corrosion behavior than 8001 aluminum alloy. A tentative process for fabricating hot die size fuel clad with high-silicon aluminum has been established.

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TABLE X

SUMMARY OF HOT DIE SIZE FUEL
DIMENSIONAL CHANGES DURING IRRADIATION

	<u>Initial Test</u> <u>PT-IP-546-A</u>		<u>End Bond</u> <u>Test-Induction</u> <u>vs. Resistance</u> <u>PT-IP-665-A</u>			<u>Parameters</u> <u>Test</u> <u>PT-IP-684-A</u>		<u>Heat Treat</u> <u>Test **</u> <u>PT-IP-708-A</u> <u>HDS</u>	
	<u>AlSi</u>	<u>HDS</u>	<u>AlSi</u>	<u>Resist.</u>	<u>Induc.</u>	<u>AlSi</u>	<u>HDS</u>	<u>Water</u>	<u>Oil</u>
OD (Ave. Ends)-Mils	7.9	11.0	4.8	10.1	8.9	6.2	9.4	7.0	2.1
OD (Center)- Mils	9.0*	16.6	5.8*	14.5	12.5	7.9*	13.1	12.1	3.4
ID (Ave.)- Mils	4.5	10.0	2.9	12.9	12.4	5.4	6.8	13.3	0.1
Wt. Loss- Grams	1.6	2.0	1.2	1.4	1.2	---	--	--	---
MWD/T		~ 960		835			985	--	
Specific Power KW/Ft.		~1125		~1080			1100	--	

* The over-all dimensional change in center diameter for AlSi fuel in all reactors as measured by Quality Certification charges varies from -10 to +20 mils.

** Because growth of HDS fuel established, no AlSi control pieces prepared for PT-IP-708-A. AlSi pretreatment and Alpha anneal of test material had no significant effect on dimensional behavior during irradiation.

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LEGEND:

PROCESS	CORE TREATMENT	
	WATER QUENCH	OIL QUENCH
ALSI BRAZE		—
HDS RESISTANCE		—
HDS INDUCTION		

FIGURE 3 - INTERNAL DIAMETER CHANGES AT THE MIDPOINT OF HOT DIE SIZE FUEL DURING FOUR PRODUCTION TESTS

TEST NO.	DESCRIPTION
546	Initial Hot Die Size Irradiation
655	Induction vs. Resistance Heated End Bonding
684	Parameters Test
708	Heat Treatment-Oil vs. Water Quench

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LEGEND:

PROCESS	CORE TREATMENT	
	WATER QUENCH	OIL QUENCH
ALSI BRAZE		-
HDS RESISTANCE		-
HDS INDUCTION		

FIGURE 4 - EXTERNAL DIAMETER CHANGES AT THE MIDPOINT OF HOT DIE SIZE FUEL DURING IRRADIATION OF FOUR PRODUCTION TESTS

TEST NO.	DESCRIPTION
546	Initial Hot Die Size Irradiation
655	Induction vs. Resistance Heated End Bonding
684	Parameters Test
708	Heat Treatment-Oil vs. Water Quench

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DATE

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8/15/94

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