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FUEL ELEMENT DEVELOPMENT COMMITTEE

ANNUAL REPORT FROM HANFORD

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AUTHOR

M. Lewis-NRD

J. E. Minor-BNW

J. T. Stringer-IPD

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FUEL ELEMENT DEVELOPMENT COMMITTEE

ANNUAL REPORT FROM HANFORD

by:

M. Lewis - NRD
J. E. Minor - BNW
J. T. Stringer - IPD

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August 18, 1965

HANFORD ATOMIC PRODUCTS OPERATION
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SECTION I

SUMMARY OF HAPO ACTIVITIES

I. K AND SMALL REACTORS (Excluding N-Reactor)

Over-all performance of fuel in the small and K Reactors for FY 1965 was essentially unchanged from the preceeding year. In neither year did fuel element quality limit reactor operation or the attainment of reactor production goals. K Reactor self-supported fuels sustained five reactor failures, three of which were caused by brittle end bonds. Fifty-four (54) failures occurred in non-bumper fuel in the small reactors. Seventy-five (75) per cent of these failures were side hot spot accelerated corrosion-type failures which the use of bumpers will diminish. Bumpered fuel in the small reactors sustained eleven failures, three of which were manufacturing defects, three were groove corrosion, and five were unclassified or unknown.

Measurable attributes which are believed to influence fuel element performance did not change significantly during the year. Irradiation-induced warp trended upward from an average of about four mils to an average of about five mils. Total external bond count showed a slight improvement over the previous fiscal year.

For the small and K Reactors, emphasis during the year was on (1) maintaining the existing quality level of the AlSi fuel and achieving cost reductions in fuel unit cost, (2) continuation of the development of the hot-die-sized fuel as a replacement for the AlSi fuel with particular efforts on the solution to the problem of diameter growth which was observed in the hot-die-size fuel during early irradiation, and (3) a major effort and program to produce thoria target elements for U-233 production.

Specific programs on AlSi I&E fuel element technology are:

- (1) Continued evaluation of oil quenching versus water quenching for uranium fuel cores, which indicated that oil-quenched fuel cores for AlSi fuel continued to show less warp than water-quenched cores, but not as significant an improvement as earlier tests indicated.
- (2) Continued evaluation of induction heating instead of salt bath heating for uranium heat treatment, which continued to show no differences between the two.
- (3) Continued development and implementation of simplified supports for self-supported and bumper fuel in the small and K Reactors to achieve significant reductions in the cost of attaching rails and to achieve better performance in the reactors.

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Incentives continue to exist for increasing the length of fuel elements for the small and K Reactors, but effort on this program has been held in abeyance until the conversion to hot-die-sizing is further along.

Programs to produce approximately 25 KG of 18 per cent Pu-240 in the small and K Reactors continued during the year. Depleted uranium has been irradiated in B Reactor and in KE Reactor to exposures of approximately 1950 MWD/T and 2200 MWD/T without incident. Tentative plans are to continue irradiation to approximately 2500 MWD/T before discharging.

During the fiscal year, the production of U-233 has grown from a program to irradiate six tons of thoria to a program where 153 tons of thoria have been discharged from the small and K Reactors as both fringe and core loadings. Problems have been encountered in the quality of the ThO₂ received, and these have not yet been adequately resolved. Specifications have been established by concurrence of the four production sites involved which, if followed, produces a barely adequate product for compacting into target elements of the desired density and which will dissolve readily in the separations plants. During the irradiation of 90 tons of oxide, 21 thoria elements failed by water entry through weld defects. When this was recognized as a serious problem, improved target element testing procedures were instituted which included radiography and helium leak testing. Subsequently, 63 tons of thoria target elements were irradiated with only one failure.

During the initial irradiation testing of hot-die-sized fuel elements, it was observed that the center OD of the hot-die-sized fuel elements grew approximately eight mils more than the equivalent AlSi control pieces. Both hot-die-sized and AlSi pieces were fabricated from uranium which had been salt-water quenched heat treated. A number of tests were carried out to define the problem and the solution was found to be in oil quenching the uranium cores rather than water quenching them. Consequently, specifications for uranium cores for the hot-die-sized fuel elements will include salt-oil quenching in the future. A pilot line with a capacity of approximately 5,000 hot-die-sized fuel elements per month was placed in operation during the fiscal year, and it currently produces fuel for the small and K Reactors at approximately that rate.

II. N-REACTOR

A total of 907 tons of production fuel elements, comprising the first two reactor loads, has been manufactured. This includes 88 tons of spike enrichment fuel in which the outer tube is 1.25 per cent U-235. Manufacturing yields have continued to improve and costs to decrease. Irradiation performance has continued to demonstrate the validity of N-fuel design, materials, and manufacturing processes. In addition to the standard fuels, 17.5 tons of coproduct driver test fuels have been manufactured. Some of these have been made from higher enriched uranium and some from developmental alloys. Irradiations of these fuels are partly completed.

Average fuel exposure in the reactor is 1600 MWD/T with a maximum piece exposure of 2755 MWD/T. Four fuel element failures have occurred. Analyses

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of the failures show no weaknesses in fuel design, materials, or manufacture. The rupture monitoring system has functioned promptly. Contamination has not interfered excessively with operation or maintenance. Two hundred tons of the first load will be exposed to 2100 MWD/T.

Development programs and process improvements are underway in the following areas:

- (1) Uranium billet fabrication
 - (a) Tipped mandrel extrusion
 - (b) Ingot piercing
 - (c) Upset forging
 - (d) Direct casting
 - (e) Specification revisions
- (2) Alternate uranium composition
 - (a) Fabrication development
 - (b) Capsule irradiations
 - (c) Full size fuel element irradiations
- (3) End closure development
- (4) Improved material utilization
- (5) Advanced fuel element designs

Each of these topics is discussed in the text of the report.

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SECTION II

K AND SMALL REACTOR PRODUCTION FUEL

I. FUEL PERFORMANCE

Average fuel quality as measured by the relative rupture resistance of non-bumper fuel at the small reactors was constant throughout FY 1965 and was at the same level as during the last quarter of FY 1964. Table I below summarizes the fuel performance data over the past two years. Irradiation-induced warp and external bond count are the most meaningful measurements which relate to fuel performance, and the rupture resistance index is a measure of fuel quality based on rupture frequency.

TABLE I

PERFORMANCE INDICES, NON-BUMPER I&E FUEL
B, D, DR, F, H REACTORS

Third Quarter 1962 = 100 - Decreasing Indices Indicate Improvement

<u>Discharge Period</u>	<u>Natural Only</u>			<u>Enriched</u>
	<u>Warp</u>	<u>Bond Count</u>	<u>Rupture Resistance</u>	<u>Rupture Resistance</u>
	<u>Index</u>	<u>Index</u>	<u>Index</u>	<u>Index</u>
3rd Quarter CY 1963	7	60	17	8
4th Quarter CY 1963	76	80	50	12
1st Quarter CY 1964	51	102	50	23
2nd Quarter CY 1964	41	148	100	46
3rd Quarter CY 1964	52	124	100	46
4th Quarter CY 1964	68	104	100	46
1st Quarter CY 1965	79	116	100	46
2nd Quarter CY 1965		90	100	46

The above data is only for non-bumpered fuel in the small reactors in which 54 failures occurred during the year. Forty of the 54 failures were side hot spot accelerated corrosion failures which the use of bumpers will diminish. Conversion to bumper fuel is in progress.

The performance of bumpered fuel in the small reactors has been good. Eleven failures occurred during the fiscal year in bumpered fuel. Three

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were manufacturing defects, three were groove corrosion, and five were unclassified or unknown.

Only five failures occurred during the year in self-supported fuel in the K Reactors, one in natural and four in enriched elements. Three of the failures were caused by brittle end bonds, one by localized ledge corrosion, and one by mechanical damage.

II. SPECIFIC PROGRAMS

1. Uranium Specifications

New specifications for I&E cores were issued during the fiscal year. The major changes were in establishing minimum limits of 75 ppm Si and 100 ppm Fe. Previous specifications had maximum limits of 140 ppm Si and 200 ppm Fe and a minimum of 140 ppm Fe plus Si. The maximum limits were not changed in the new specifications.

2. Projection Fuel Program

The bridge-type self-support rails for fuel loaded in ribless tubes have been replaced with "arch" self supports. The arch self support has the advantage of lower cost, a better load-deflection curve, lower hydraulic resistance, and only one ultrasonic weld is required on each rail. Conversion from the bridge to the arch self support was completed in July 1965 for self-supported fuel. Conversion to a modified type arch bumper support for use in ribbed tubes is in progress.

3. Overbore Program

During FY 1965, seven failures occurred in the C-size overbore fuel elements. All of the failures were caused by poor bonding on the ends of the fuel, and all occurred in AlSi-brazed elements. Conversion from AlSi to hot-die-sizing on the overbore fuels was started in the second quarter of CY 1965, and 60 per cent of the overbore elements now in C Reactor are hot-die-sized. The stronger end bonds on hot-die-sized elements should improve the performance of the overbore fuel. Current plans are to clad all overbore fuel by hot-die-sizing.

4. Salt-Oil Heat Treating (AlSi Cores)

Additional data developed through the year continues to support the improved dimensional stability of salt-oil heat-treated cores, but the improvement over salt-water heat treating is not as great as earlier data indicated. A 15-column test comparing salt-water and salt-oil heat-treated enriched O3E cores showed the same warp on the two types of cores but a lower ΔOD_2 and a lower tube filling capacity on the salt-oil heat-treated cores. Data from the bulk charging showed a lower warp than the average for salt-oil, but the values were within the spread normally observed with salt-water. Salt-oil heat treating has significant advantages over salt-water for the hot-die-sizing program, and this is discussed in the section dealing with alternate process development.

5. Induction Heat Treating

The bulk charging of induction-oil heat-treated O3N cores has continued at the rate of approximately 20 tons per month, and a total of 157 tons

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has been charged. The performance has been satisfactory, and the post-irradiation data show a lower than normal diametral growth and warp at the same level as salt-oil cores.

6. Increased Fuel Element Length

A preliminary production test comparing six-inch and ten-inch salt-water heat-treated and AlSi-canned cores showed no significant difference in irradiation-induced warp between the two lengths. There was a statistically but probably not operationally significant difference in ΔOD_3 , ΔID_1 , ΔID_2 , and ΔID_3 , with the ten-inch fuel exhibiting greater diametral changes on all four attributes. Further work on long fuel is in abeyance pending developments in the hot-die-sizing program and the development of more economical techniques for producing longer cores at the feed site.

7. Depleted Uranium

Approximately 12.5 tons of 0.22 per cent U-235 cores were charged in B Reactor during July 1965, and 15 tons were charged in KE Reactor during August 1964. The objective of the program is to produce 25 KG of 18 per cent Pu-240 plutonium. To achieve this, average exposures of 1200 MWD/T and 1900 MWD/T are required in the B and K Reactors, respectively. The irradiations have proceeded without incident, and the average exposures are now at 1950 MWD/T in B and 2200 MWD/T in K. Tentative plans are to irradiate both loads to 2500 MWD/T. Test columns have been discharged at various exposures to monitor the behavior of the depleted elements. The results are shown below:

Reactor	B	KE	
Exposure (MWD/T)	775	750	1475
ΔOD_1 (mils)	1.8	- 1.7	5.7
ΔOD_2 (mils)	3.8	- 0.5	7.3
ΔOD_3 (mils)	4.8	- 3.3	2.0
Warp (mils)	6.0	10.6	2.7
TFC	11.0	12.0	11.3

Corrosion on the K elements has been limited to slight pitting less than 10 mils deep.

8. U-233 Program

During fiscal year 1965, the U-233 program has grown from a six-ton demonstration load to a program where 153 tons of thorium have been charged in the small and K Reactors. The elements are prepared by vibratory compaction of sol-gel type thorium in approximately 1-1/2 inch diameter aluminum cans. The vibration time varies from 30 seconds to three minutes, depending upon the characteristics of the oxide. The target core density has been 7.5 g/cc, but this has been difficult to achieve and many of the elements have been in the 6.9 to 7.3 g/cc range. Several problems have been encountered in the quality of the thorium oxide received, and

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these have not yet been adequately resolved. The principle problems have been in the compactability and hygroscopic nature of the oxide. The compactability is dependent largely upon two characteristics of the oxide; particle size distribution and particle density. Thoria has been received from two producers, Nuclear Fuels Services and Mallinckrodt Chemical Works. In general, the NFS oxide has been superior to the MCW oxide with respect to compaction. A comparison of average particle size distributions, particle densities, and compaction densities is shown below:

<u>Sieve Analysis (%)</u>	<u>MCW</u>	<u>NFS</u>
+ 6 mesh		0.0
- 6 + 20 mesh	3.5	12.5
- 20 + 70 mesh	25.5	27.2
- 70 +200 mesh	20.6	21.0
-200	50.1	39.4
Particle Density (g/cc)	9.1	9.4
Compaction Density (g/cc)	7.0	7.4

The data above is on MCW lots produced from refined thorium nitrate solution and on NFS lots produced from unrefined thorium nitrate tetrahydrate.

Another problem has been the tendency for thoria to absorb moisture from the atmosphere. This has been an intermittent problem on both MCW and NFS lots and is apparently related to the quality of the starting material and the conditions of high-firing. Moisture contents of 0.18 per cent have been observed on canned thoria. The effect of moisture during irradiation has not yet been determined.

Twenty-one (21) thoria elements failed under irradiation in the first 90 tons of oxide irradiated to goal exposure. All of the failures examined were apparently caused by water entry through weld defects. Improved weld testing techniques were instituted when it became apparent there was a serious problem, and in 63 tons of irradiated elements, inspected under improved methods, only one failure occurred. The improved inspection includes radiography and helium leak testing. Failures generally have been characterized by either a uniform swelling of the element or a bulging near the cap end. The former appears to be caused by actual swelling of the thorium oxide.

Canning of an additional 53 tons for reactor fringe loading has recently been completed.

9. Alternate Process Development

Development work progressed on the hot-die-size process as a replacement for ALSi canning. Early irradiation testing showed the hot-die-sized elements had a five to ten mil greater diametral growth on both the OD and the ID than did the ALSi elements. Efforts during FY 1965 have been directed toward reducing the diametral growth, equipping and operating a

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pilot line with a 5,000 element per month capacity, and making process improvements.

A summary of completed hot-die-size irradiation tests is presented in Table II. Most of the tests were directed toward reducing the diametral growth of hot-die-sized cores. A highly significant improvement was achieved in eliminating growth by oil quenching rather than water quenching the fuel cores during heat treatment. A summary of the results of this test is given below:

	Diametral Growth	
	Salt-Water Cores	Salt-Oil Cores
ΔOD_1 mils (end)	5.79	1.48
ΔOD_2 mils (center)	12.07	3.43
ΔOD_3 mils (end)	2.73	8.29
ΔID_2 mils (average)	0.14	13.30

Based upon the above results, the specifications for hot-die-sized fuel cores were revised to require salt-oil heat treating of all core models.

Testing was also carried out to determine the performance of fuel made with tube cladding rather than impact extrusions. Canning and irradiation were successful and the conversion to tube cladding on all except the C-overbore model will begin in August 1965. Economics strongly favor the use of tubing components.

During the year, a pilot plant was equipped and a capacity of 5,000 HDS elements per month has been demonstrated. Process development was directed toward process optimization and improvement. Major changes were the replacement of resistance heat with induction heating for end bonding, increasing the ram speed through the sizing dies by a factor of three, to 240 inches per minute to reduce cladding thickness variation and improve end bonding, and the use of slightly tighter fitting end caps to reduce the closure weld reject rate from 25 per cent to less than one per cent. Also, complete conversion from AlSi brazing to hot-die-sizing of C Reactor overbore fuel was accomplished during the year.

Several tests are in progress to compare the quality of hot-die-sized and AlSi brazed fuel, and to determine the optimum heat treatment and core composition for hot-die-sized fuel. The quality comparison tests consist of a 0.95 enriched test in a small reactor in which alternate tubes of AlSi and hot-die-sized fuel will be irradiated until a set number of ruptures occur, and a 1.25 enriched test in a K Reactor in which the cladding corrosion and dimensional stability of hot-die-sized and AlSi fuel under more severe operating conditions will be compared. A core optimization test is also being prepared in which cores with solid salt-water, hollow salt-water, hollow salt-oil, and hollow induction-oil heat treatments and iron, silicon, and aluminum levels at 50, 175, and 300 ppm will be irradiated. Also, a standard alloy test of solid salt-oil and solid induction-oil heat treatments

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is being prepared. The results of these tests will provide answers to the optimum composition and heat treatment required for best performance of hot-die-sized fuel. Tables II and III summarize the irradiation tests performed or in progress, and Figure 1 shows the bulk charging schedule.

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TABLE II

SUMMARY OF IRRADIATION TESTS COMPLETED
HOT DIE SIZING

Test	Charged	Discharged	Purpose of Test	Results
PT-IP-546-A 20 Columns HDS & AlSi Control Fuels (Normal U)	7-27-63	12-4-64 (120% of goal)	Initial irradiation of normal U HDS fuel. Evaluate dimensional stability. Evaluate corrosion behavior of cladding.	Above-normal discharge force req'd. Increase in OD up to ~ 0.008" greater than companion AlSi pieces. Average exposure ~ 950 MWD/T.
PT-IP-616-A 19 Columns HDS & AlSi Control Fuels (0.95 Enriched U)	9-28-63 (13 cols) 10-30-63 (6 cols)	12-4-64	Initial irradiation of enriched U HDS fuel. Evaluate dimensional stability. Evaluate corrosion behavior. Further evaluation of HDS process.	Above-normal discharge force req'd. Dimensional behavior comparable to PT-IP-546-A Average exposure ~ 900 MWD/T.
PT-IP-546-A Supplement A 10 Columns HDS & AlSi Control Fuels (Normal U)	2-27-64	4-28-64	Obtain data in coolant tempera- ture distribution vs. fuel dimensional behavior.	Nine of ten tubes required above- normal discharge pressures. Exposure ~ 540 MWD/T.
PT-IP-665-A 12 Columns HDS & AlSi Control Pieces (Normal U)	4-29-64	9-1-64	Evaluate effects of resistance heat vs. induction heat end bond- ing on in-reactor dimensional behavior.	Discharged from C Reactor 9-1-64. Normal force required for dis- charge. Slight reduction in growth of induction heated fuel. Exposure ~ 800 MWD/T.
PT-IP-677-A 12 Columns HDS Fuels (Normal U)	12-22-64	3-25-65	Evaluate the use of tube cladd- ing components in the HDS process.	No problems associated with tube cladding. Discharge required higher than normal forces.
PT-IP-684-A 24 Columns HDS & AlSi Control Fuels (Normal U)	7-16-64	11-13-64	Evaluate "skin" effect of heat treating on cores. Evaluate pressure variables in end-bonding operation.	High discharge forces required in six columns. No significant dif- ference in growth between AlSi and HDS test fuel.
PT-IP-708-A 18 Columns HDS Fuels (Normal U)	9-1-64	12-21-64	Evaluate influence of the AlSi process on in-reactor dimensional behavior of HDS fuels. Evaluate effects of oil-quench and water-quench in core heat treatment.	Normal discharge pressure. Sig- nificant difference between water- quench and oil-quench HDS fuel. Oil quench better by factor of 3-4. Exposure ~ 800 MWD/T.

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Total Charged Through 12-21-64 = 115 columns of test material in 12-21-64

TABLE III

SUMMARY OF IRRADIATION TESTS IN PROGRESS
HOT DIE SIZING

<u>Test</u>	<u>Charged</u>	<u>Scheduled Discharge</u>	<u>Purpose of Test</u>
PT-IP-781-A Run-to-Rupture 408 HDS-408 AlSi Fuels 0.947 Enriched D Reactor	7-19-65	~ 12/15/65	Compare irradiation performance of AlSi and HDS process fuel under similar reactor conditions.
PT-IP-797-A Equivalency Test 195 HDS-195 AlSi Elements 1.25 Enriched K Reactor	~ 9-15-65	~ 11-30-65	Compare corrosion characteristics and performance of AlSi and HDS fuel under severe reactor conditions.
Core Optimization Test ~ 1600 HDS Elements Natural Uranium C Reactor	~ 10-16-65 & 11-16-65	~ 12/15/65 & 2/15/66	Evaluate behavior of HDS cores with varying amounts of Fe, Si, and Al (up to 300 ppm each) on five heat treatments: solid salt-water, hollow salt-water, hollow salt-salt, hollow salt-oil, hollow induction-oil.
High Silicon Cladding ~ 200 HDS Elements 1.25 Enriched	~ 12/15/65		To compare corrosion rates of X-8001 and high silicon (~ 10%) alloy cladding.
Heat Treatment Test 320 HDS Elements Natural Uranium	~ 1/15/66	~ 4/15/66	Compare behavior of hollow and solid induction-oil and salt-oil heat treatments.

FIGURE 1

SCHEDULE FOR BULK CHARGING
OF HOT-DIE-SIZE FUELS

SECTION III

N-REACTOR FUELS

I. PRODUCTION STATUS AND FUEL PERFORMANCE

A. Fuel Element Manufacturing

The manufacturing and irradiation performance of the N-Reactor fuel element has continued to demonstrate the validity of the design, materials, and processes. Manufacturing yields have continued to improve and costs to decrease.

The initial quantity of fuels for the first load consisted of 372 tons of 0.947 per cent U-235 enriched tube-in-tube fuel elements.

Since fabrication of the first load, an additional 447 tons of 0.947 per cent U-235 enriched tube-tube fuel elements have been produced together with 88 tons of spike enrichment with an outer tube of 1.25 per cent U-235 enrichment and an inner tube of 0.947 per cent U-235 enrichment. No major design changes in the fuel or manufacturing process were necessary for this second load of fuels and the manufacturing process has remained essentially constant. A pictorial flow sheet is shown in Figure A.

In other terms, the completion of manufacturing of two reactor loads means that:

- 6,000 coextrusions have been made,
- 40,000 fuel assemblies have been completed and certified,
- 160,000 fuel element closures have been made,
- 700,000 pieces of support hardware have been attached to fuel element cladding with 1,400,000 spot welds,
- Approximately 20,000,000 square inches of nominally 20 mil zircaloy-2 cladding have been examined for minute defects and for bonding to uranium and have been found defect-free.

These statistics serve to amplify the soundness of the technology developed for N-fuels fabrication. This is further demonstrated by the ability to produce fuels of various new geometries and using different uranium alloys as discussed further below.

In addition to the standard fuels, 11 tons (38 reactor tube charges) of 1.25 per cent U-235 enriched coproduct driver test fuels containing a target rod of lithium-aluminum alloy have been irradiated. Six and one-

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CO-EXTRUDED FUE

L FLOW SHEET

FIGURE A

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Page 17

ATED BILLET
SEMBLY

ER JACKET
SEMBLY

PSURE

BRIGHT ETCH

COPPER
ZIRCALOY-2
URANIUM
ZIRCALOY-2
COPPER
MANDREL

BILLET
PREHEAT

CO-EXTRUSION

FORCE
MANDREL
RAM
DIE
BILLET

CUT LENGTHS
AND REMOVE
SCRAP ENDS

HNO₃ COPPER
REMOVAL

END RECESSING - CHEMICAL MILLING

OUTER AND INNER
FUEL ELEMENTS

AUTOCLAVE

FINAL ASSEMBLY

half tons (18 reactor tube charges) of 1.96 per cent U-235 enriched coproduct test fuels are being irradiated. About 25 per cent of the fuels in this second group were fabricated from a developmental uranium alloy containing 350 ppm iron and 800 ppm aluminum. Four reactor tubes of 0.947 per cent U-235 enrichment, but fabricated to the coproduct geometry from another developmental alloy of uranium containing 350 ppm iron, 350 ppm silicon, 800 ppm aluminum are under irradiation. Currently, an additional 14 tons (approximately 40 reactor tube charges) of these simulated driver fuels in various alloys are being processed for irradiation.

Conversion costs for the fuel manufacturing process have continued to show the predicted substantial improvements as shown in Table I. These gains in unit conversion costs have resulted from improved manufacturing yields both in terms of total material utilization and piece yields as tabulated in Table II. Further contributions to reduce unit costs have been attained through significant reductions in the price paid for the zircaloy components. A 24 per cent reduction in the average price of zircaloy components was gained during the year compared with fiscal year 1964, for a total saving of about \$450,000. These cost reductions for zircaloy components have been gained through better purchasing practices and specification improvements while the basic price of the zircaloy ingot increased about 20 per cent. Future gains in materials costs will be derived from improved materials utilization; further reductions in basic zircaloy prices are unlikely.

B. Fuel Element Performance

The first full N-Reactor load has performed in accordance with the most optimistic expectations. On August 1, 1965, average exposure reached 1600 MWD/T, with a maximum column exposure of 1815 MWD/T and a maximum fuel element exposure of 2755 MWD/T. Visual examinations of some 200 discharged fuel elements shows them to be in generally excellent condition even though these fuels have had a 20 months' exposure in the reactor compared with a planned four to six months. They have undergone about 35 rapid thermal cycles from reactor scrams compared with five to ten expected for future loads. All design parameters such as swelling, structural integrity, absence of deleterious vibrations, suitability for charging and discharging operations appear fully confirmed.

Four fuel element failures occurred in the first load through the end of FY 1965. One of these occurred shortly after startup and was described in last year's report. The other three are described and analyzed briefly below. Analysis of the failures shows no weaknesses in fuel design, manufacture, or materials technology. The failures have also demonstrated the ability of the reactor rupture monitoring system to function promptly. Also, it has been shown that the reactor system is able to tolerate and handle nominal fuel failures without dire consequences from contamination, downtime, and operating costs. Data to date indicate that the fuel is fully capable of exposure to at least 2100 MWD/T. Authorization has been provided to irradiate 200 tons of the charge to that exposure and this is underway.

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TABLE I

N-Fuels Manufacturing Costs

First Load Average	\$10.00/lb. for 372 tons total
FY-1963 Average	10.65/lb. for 209 tons
FY-1964 Average	7.88/lb. for 342 tons
FY-1965 Average - Actual	6.45/lb. for 408 tons
- Target	6.43/lb. for 420 tons

TABLE II

Summary of Piece and Weight Yields

Major Losses (Piece Yields)	FY-1965			First Load		
	Outer	Inner	Total	Outer	Inner	Total
Coextrusion Bond	99.4%	98.2%	98.8%	96.1%	92.1%	96.6%
Extrusion Surface	99.0%	99.5%	99.2%	99.8%	99.9%	99.8%
Warp	99.7%	89.8%	94.5%	99.8%	97.3%	98.5%
Clad Thickness	92.4%	97.7%	94.7%	87.0%	99.0%	93.0%
End Closure (partially recoverable)	98.3%	98.9%	98.6%	87.0%	92.7%	89.9%
Uranium (weight) Utilization	74.3%	77.1%	75.7%	42.0%	56.7%	46.6%

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A large amount of charging and discharging has been performed without difficulty. The support system, especially with the new 50 mil thick supports, appears to be adequate for current needs. Experimental measurements of coolant temperatures in each of the three coolant passages are being made with thermocouple trains behind a number of fuel columns in the reactor. In general, these measurements confirm predictions with regard to flow distribution and heat generation. Evaluation of the data will provide a basis for any minor dimensional changes that might be made to improve water flow balance to increase power output capability.

C. Fuel Failure Analysis

Four failed fuel elements have been identified since startup, one in each of four columns discharged because of fission product activities in the coolant. A brief summary of the four failed elements is given in Table III.

Failure No. 1 was described in last year's report and the description will not be repeated. None of the more recent failures is of this type, which may be described as a "delayed autoclave failure."

TABLE III

Summary of N-Fuel Failure Information

<u>Failure Number</u>	<u>Date</u>	<u>Approximate Fuel Element Exposure (MWD/T)</u>	<u>Remarks</u>
1	7/ 6/64	10	An extrusion defect through the 0.220 inch thick end cap permitted water entry to the inner fuel element.
2	5/15/65	400	Standard fuel elements. Water entry was through the inner surface of the outer element in the vicinity of (but not under) a buggy-spring support.
3	6/ 6/65	800	
4	6/15/65	1600	Coproduct production test fuel element. Water entry appears to be under a locking clip attached to the inner surface of the driver element.

Failure No. 2 occurred through the inner clad of an outer component immediately adjacent to the contact area of an inner component buggy-spring support. The site of initial water entry has been identified as a highly localized thin spot or defect in the clad. Detailed metallography has not revealed any direct evidence that the defect was an inclusion, seam, lap, crack, or tear, although this possibility cannot be completely

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ruled out. Several features of the defect strongly suggest that failure resulted from fretting corrosion but not from the buggy spring itself. The occurrence of fretting corrosion would then require the entrapment of some foreign material by the buggy spring support and a subsequent fretting penetration of the clad by the foreign material. Some evidence that a foreign object was present has been found.

Failure No. 3 also occurred through the inner clad of an outer component with the damage involving the contact areas of two of the three center inner component buggy spring supports. The site of initial water entry bears a striking resemblance in appearance and location to that of the second failure. Examination thus far indicates that fretting corrosion by entrapment of some foreign material on the buggy spring offers the best explanation of this failure. A foreign object, believed to be a small irregular shaving of chromium plating, has been discovered on the buggy spring and appears to have been the fretting member.

Failure No. 4 occurred in the inner bore of the driver element of a co-product assembly. Extensive fuel corrosion and undercutting of the clad occurred before clad fracture and the release of fission products and corroded fuel. The generalized nature of the damage is such as to preclude an accurate estimate of the initial site of failure at this time. However, it appears to be under a locking clip, although it is not yet certain how this fact contributes to the cause of failure.

D. Experimental Irradiation Experience

1. N-Prototype Tests

The irradiation testing of 13 prototype coproduct fuel-target assemblies was completed in July 1965 in a KER facility to demonstrate and characterize the simultaneous production of plutonium and tritium. These assemblies were irradiated to 1270 MWD/T tube average in 1350 psi coolant at an average driver element power of 106 KW/ft. The peak exposure in the test charge was 2220 MWD/T attained at a peak power of 185 KW/ft. Maximum fuel swelling in the column was 3.32 per cent and the element that attained this swelling at times operated with maximum fuel temperatures up to 540 C. Standard N-fuel composition, 150 Fe - 100 Si, was used in these tests. Fuel metallography indicated that swelling resulted from the development of "grain boundary tearing" in the lower temperature metal in the fuel and crystallographically aligned micro-porosity in the higher temperature metal. Measurements indicate good dimensional stability with a maximum outer radius increase of 0.005 inch and a maximum inner radius decrease of 0.010 inch. There was no indication of fuel element warp or inner bore buckling in the test charge.

2. Comparative Swelling Behavior of Uranium Alloy Fuel Elements

Tubular fuel elements (KSE-5's) containing uranium with Fe and Si additions (140 ppm and 120 ppm, respectively) and uranium with Fe

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and Al additions (400 ppm and 800 ppm, respectively) have been irradiation tested in high-temperature, high-pressure KER loops in two tests reaching 1530 MWD/T average and 2700 MWD/T average exposures.

The test charges were loaded so that the two fuel compositions occupied alternate positions in the columns and thus each composition experienced similar gradients of operating temperatures and exposures with identical irradiation histories. The uranium fuel containing Fe and Al additions swelled about as much as might be anticipated to accommodate the solid fission products (three v/o per one a/o B.U.), while the uranium containing Fe and Si additions swelled in the maximum case about 2-1/2 times more than this theoretical minimum. Post-irradiation metallographic studies reveal a small amount of "grain boundary tearing" in both fuel compositions in the cooler regions (~ 400 to 450 C), apparently a consequence of the anisotropic growth behavior of alpha phase uranium. In the hotter regions of the fuel (~ 500 to 575 C), however, there is a distinct difference in the swelling behaviors. The Fe and Si-bearing fuel displays clusters of oriented porosity whereas the Fe and Al-bearing fuel is relatively free of such porosity. Post-irradiation fuel density determinations corroborate the swelling observations made on the basis of fuel element bulk density changes and fuel metallography. It is believed that the swelling in both cases is the result of mechanical damage in the alpha phase uranium from the irradiation environment and the fission process and is not a consequence of the agglomeration of fission product gases. It is possible that the degree and/or type of mechanical damage observed in the fuel may be related to the recrystallization or recovery kinetics of the uranium as influenced by the alloying additions.

3. ETR High Exposure Test of Fuel Material Containing Fe-Al

A test charge of three eight-inch long tubular fuel elements containing uranium fuel with 400 ppm Fe and 800 ppm Al additions is being prepared for irradiation testing in the ETR M-3 loop. Goal exposure is 5000 MWD/T. The elements are 1.79 inch OD, 0.97 inch ID, coextrusion clad with 0.025 inch of Zr-2. End closure rings of Zr-2 are brazed in place with Be-Zr eutectic. The elements are intended to operate at 160 KW/ft with maximum fuel temperature of about 600 C. Fuel swelling data will be obtained by fuel element density measurements made at the end of each ETR cycle.

E. Fuel Examination Facility

The project is almost complete to provide special discharging and handling equipment and facilities for post-irradiation measurements for research and development fuel from N-Reactor. The special handling equipment will permit discharging special fuel assemblies so as to preserve their as-irradiated condition rather than to discharge them by free-fall into the discharge pit, as is done with production fuel. The measuring equipment is being located in the N-Reactor storage area using the water

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there for shielding. It will consist of a radiation measuring device for relative exposure determinations, a separator for inners and outers, visual inspection station, a weighing station, warp and length gages, a cladding thickness tester, automatic dimensional gages, and end profilometer and other equipment needed to accomplish the post-irradiation characterization of the fuel. Some of the equipment that was used in the facility associated with the KER test loops is being upgraded and used in the new facility.

Examinations have been performed in this facility of the four fuel columns discharged for rupture indications. Observations of unusual or otherwise interesting features will be confirmed in the radiometallurgy laboratory. Weighed fuel elements from monitor columns and other production test fuels have been reweighed in the facility. Swelling calculations from these weighing data indicate close agreement with data previously obtained from KER loops. Dimensional measurements will not be possible until completion of the special handling features now scheduled for early 1966.

II. DEVELOPMENT PROGRAMS AND PROCESS TECHNOLOGY

A. General

Development work in the N-fuels program is aimed at the following objectives.

1. Developing the technology, fuel element designs, and fuel fabrication processes for potential operating modes and products from N-Reactor that take full advantage of the design margins of N-Reactor and the N-fuel for either coproduct production, higher power levels, and/or longer exposures.
2. Improving the quality of the fuel element by development of designs that include additional margins for reliable operation at increased power and exposure and by improving uniformity of the end product.
3. Improving the quality of the fuel elements by improving processing methods and quality control techniques.
4. Reducing manufacturing costs through:
 - a. Yield improvements, as a result of tighter process control and process changes;
 - b. Design changes which allow reductions in the use of process materials;
 - c. Reduce costs of purchased materials through improved specifications, definitions of quality, development of new sources.

B. Uranium Billet Fabrication

The uranium billet core is presently manufactured by casting an 11-inch diameter by 30 inch long solid ingot containing a nominal 150 ppm iron

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and a nominal 100 ppm silicon. The ingot is drilled, triple beta heat treated, and primary extruded into raw billet stock. The dimensions of the primary extrusion require a machine removal of about 0.25 inch of uranium on the OD and about 0.20 inch on the ID. Material design utilization from ingot to finished billet is approximately 60 per cent.

The objectives of the uranium billet fabrication programs are:

1. Improve the structure of the uranium billets in order to reduce zircaloy cladding thickness variation. The major billet-related problem in the present process has been zircaloy-2 cladding thickness variation that is caused by local variation in resistance to deformation of the uranium, resulting in a large part from the residual cast ingot structure. The bulk of the cladding-related rejects listed in Table II are from this cause.
2. Reduce the costs of the uranium billets by simplifying processes and improving yields. These programs involve improving both the design utilization of uranium as well as reducing the reject rate for billets.
3. Develop and prove out billet designs, alloys, and billet fabrication processes needed for new N-Reactor fuels required for the more advanced operating modes of the reactor.

C. Primary Extrusion Over a Tipped Mandrel

The tipped mandrel process for billet fabrication has been developed primarily to improve the structure of the uranium billet and therefore reduce zircaloy-2 cladding thickness variation. A 13 inch OD x 4.7 inch ID hollow ingot is cast, heat treated, and extruded to a 7.2 inch OD x 2.6 inch ID and machined to the finished billet size of 6.950 inch OD x 2.800 inch ID. The larger ingot ID provides a more effective heat treat quench response because of the larger surface-to-volume ratio than in the standard ingot, with a 2.7 inch hole. The increased working of the uranium metal, particularly on the ID, by extruding over the tipped mandrel also aids in obliterating any cast ingot structures. About 64 ingots have been extruded over tipped mandrels. The coextrusion results have substantiated the anticipated improvements in cladding thickness variation. Reductions in variation on the order of two mils have been obtained by utilizing these billets. Variation was reduced to a level approaching the capability of the nondestructive testers for measuring variation. The primary uranium billet fabrication difficulty that was overcome during development of this process was primary extrusion warp, which manifested itself as machined billet wall thickness variation. Actions taken to control warp included changes in the ingot end shapes, closer tolerances on primary extrusion press components, a change in die cone angles, and positioning of the tipped mandrel. During the last campaign, utilizing approximately 40 ingots, the billet yields were comparable to those obtained from the standard production route.

Development of the tipped mandrel extrusion process for standard alloy and standard geometry has been successfully completed. Conversion to

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this process has not been initiated since capital investment is required at the feed sites to modify the ingot casting facilities. Recent work on control of cladding thickness variation has indicated that a tipped mandrel process is not necessary if uranium with higher levels of alloy additives is used. Since higher alloys of uranium may be required for some of the potential operating modes of the N-Reactor, conversion of the feed sites to the tipped mandrel process is being held in abeyance.

D. Ingot Piercing

A program for piercing solid cast ingots has been initiated by Reactive Metals, Inc., Ashtabula, Ohio, as an alternate route to drilling of the cast ingots prior to heat treatment and primary extrusion. Incentives for piercing are economic, as no uranium chips or scrap are generated as in the drilling operation.

Metallographic examination of the pierced and extruded billet stock has shown a metal structure that is essentially the same as attained with the standard drilled ingot if the pierced ingot is heat treated before primary extrusion. Pierced ingots which are not heat treated show a distance plate-like structure near the ID, although this is later removed by heat treatment of the billet. Nine billets from pierced ingots have been coextruded. Maximum cladding thickness variation of the fuels was from 0.005 inch to 0.009 inch, which is in the range of fuels produced from ingots that have been drilled. Other results from these test co-extrusions confirm that no major variation can be expected. Further work is in progress at RMI to produce a larger quantity of billets in different geometries for coextrusion evaluation. Some problems have been encountered in maintaining concentricity in the piercing operation and development of the process is not complete.

E. Upset-Forging of Uranium Billets

Upset-forging is a relatively new concept for billet fabrication. The cast uranium ingot is extruded undersize. The primary extrusion is cut into short lengths and the short lengths are reinserted into the press and upset against closed dies about 1/3 of an inch on the OD and the ID to the finished billet dimensions. There is no final machining of the billet. This process can be combined with various ingot preparation methods such as tipped mandrels, piercing, etc., and has the highest potential metal utilization (about 80 per cent) because of elimination of all machining operations. About 35 billet blanks have been successfully upset-forged into billets at RMI, Ashtabula, Ohio, and the initial coextrusion evaluations of this material are scheduled for September 1965. Development problems associated with the upset-forging process are: correction of primary extrusion wall thickness variations, and correction of any surface defects emanating from the original casting. Progress has been made in each of these problem areas.

F. Direct Cast Billets

An alternate uranium billet core manufacturing process under development at the National Lead Company of Ohio for improving the uranium utilization

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is to direct cast billets and eliminate the primary extrusion process. Existing specifications for uranium billet cores require both a primary extrusion and a heat treatment to assure sufficient removal of the residual cast ingot structure for control of cladding thickness variation. However, recent work with uranium containing slightly increased levels of iron, silicon, and/or aluminum have shown that the heat treatment step alone may be effective in removing residual ingot structure for these alloys. Limited coextrusion experience to date has shown that the cladding thickness variation using alloyed direct cast billets is not significantly different than that obtained using billets from a wrought process. Currently, work is in process at NLO to determine if billets can be cast directly to finished diameters and the machining step eliminated. A flow diagram comparing the present (standard) process with the various alternates is shown in Figure B.

G. Cost Reduction Programs

The uranium billet specifications have been revised to accept the billet output of the FMPC process whenever that process is in control or whenever the extremes of the distribution are acceptable inputs for N-fuel coextrusions. This control on the central tendency of the quality distribution has been obtained by establishing Class II limits for billets at the extremes of the distributions and restricting the per cent of Class II material acceptable to the percentage generated when these processes are in control. This concept has been expanded to encompass almost all phases of billet processing as well as all dimensional requirements of the finished billet product. Within the past year, 16 new categories of Class II process limits and four new Class II dimensional limits have been added to the specifications. Since the adoption of this technique for specifications, the need for waivers has been virtually eliminated.

Additional changes in the specifications that have had a favorable impact on costs include:

1. Analyses for isotopic content and zirconium impurity are now made on composite samples from six ingots.
2. Analyses for oxygen, tin, and phosphorus have been eliminated as experience has shown that no control problems exist with these elements.
3. Ingots slightly outside of the chemical limits may be primary extruded prior to resampling and the resample obtained from the front end of the primary extrusion.
4. Metallographic examination frequency on billet samples has been reduced from one in 25 extrusions to one in 100 extrusions.
5. Specification for crucible charge makeup has been changed to permit the use of pre-melted drip crops and eliminate the need for chemical recovery of these items.

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FIGURE B

Alternate Uranium Billet Fabrication Processes

H. Alternate Uranium Compositions

Two potential operating modes for N-Reactor may require operation at uranium central metal temperatures and exposures that would be expected to cause excessive swelling of the uranium alloy currently used (150 ppm Fe, 100 ppm Si). The two operating modes and exposure conditions of the uranium for these modes are as follows:

<u>Operating Mode</u>	<u>Fuel Geometry</u>	<u>Maximum Uranium Exposure, MWD/T</u>	<u>Maximum Uranium Central Metal Temperature</u>
High Exposure Plutonium	Tube-Tube	6000	455 C
Coproduct	Driver-Target	2200	600 C

Figure C shows an autoclaved tube-tube geometry fuel element, an autoclaved uranium coproduct driver fuel, and the zircaloy-clad lithium-aluminum target rod.

Candidate alloys for these more severe operating conditions have been selected and development of billet fabrication processes, coextrusion processes, and irradiation evaluation have been started. The following uranium alloys have been selected for evaluation in full-size fuel geometries.

Uranium Alloys

<u>Iron, ppm</u>	<u>Silicon, ppm</u>	<u>Aluminum, ppm</u>
350	---	800
350	350	800
150	350	---
250	250	250
250	350	---

These materials, fabricated into Zr-2 clad fuel elements, were both beta and gamma heat treated and studies were performed to characterize the structure, recrystallization and hardness recovery temperatures were determined, the low temperature ductility was measured, behavior under cyclic stress at 400 and 500 C determined, hardness at 300 and 400 C measured, and comparative defect corrosion behavior tested. These measurements, along with analysis of potential volume per cent second phase or potential total solid solution addition, form a basis for design of additional fuel material compositions.

The bulk of the fabrication experience to date on the alloys has been obtained on the coproduct geometry which requires an outer uranium driver fuel with a wall thickness of 0.48 inch compared with a uranium wall thickness of 0.30 inch on the current N-outer fuel element. Eighteen reactor tubes have been charged with coproduct fuels (1.96 per cent enriched U-235) in this geometry. The equivalent of five of these

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FIGURE C - A Standard Tube-in-Tube Element (Left),
Standard Driver (Center), and Standard
Target (Right)

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fuel columns contain "British composition" (350 ppm Fe, 800 ppm Al), and the balance are in the standard alloy (150 ppm Fe, 100 ppm Si). Four reactor tube charges of simulated coproduct driver fuels fabricated from a uranium alloy containing 350 ppm iron, 350 ppm silicon, and 800 ppm aluminum were loaded in June 1965. The fabrication experience obtained to date with the alloyed fuels in the coproduct geometry has shown:

1. Alpha working of the uranium between casting and coextrusion as in a primary extrusion step may not be necessary to achieve good control of zircaloy-2 cladding thickness variation on finished fuel elements.
2. A beta heat treat at some stage of billet fabrication between ingot casting and coextrusion is necessary to obtain control of zircaloy-2 cladding thickness variation.
3. Heat treating and/or casting problems with the higher alloys have caused surface breaks in the uranium billets. The problem is more severe the higher the silicon content. As a result, it has been necessary to use oil quenches on some of the alloys and to heat treat the uranium in billet rather than in ingot form.
4. A larger diameter billet size for the 8.1 inch instead of the 7-1/2 inch coextrusion liner is required to achieve sufficient reduction ratio in coextrusions to assure bonding.
5. The higher alloys exhibit considerably less cladding thickness variation than the standard alloy fuels, but all other directly measurable coextrusion and post-coextrusion fabrication results from using the higher additives show no difference from the standard alloy.

I. Alloy Capsule Tests

An irradiation test of 120 specimens of 17 uranium base alloys containing small additions of Zr, Mo, Nb, Al, Si, Fe, P, and C in various conditions of heat treatment, is being conducted in NaK capsules to evaluate the irradiation performance of uranium base alloys in terms of composition, fuel fabrication history, and corrosion behavior.

Five process tubes in a Hanford reactor were charged each with eight sample capsules and one thermocouple capsule for continuous measurement of uranium temperature during the irradiation. One column of capsules has been discharged at an exposure of 2300 MWD/T for the center capsule and 1500 MWD/T for the end capsules. Maximum uranium temperatures were 600 C in the peak flux capsule and 450 C in the capsules at the ends of the column. The remaining four columns are operating at the same maximum fuel temperatures and are scheduled for goal exposure of 4000 MWD/T.

The fuel samples were coextruded with 0.030 inch thick Zr-2 cladding and are approximately 0.6 inch OD. All alloys except the one containing

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two w/o Nb plus 400 ppm Al were coextruded at conventional high alpha uranium temperatures. To obtain satisfactory clad rods of the U + Nb + Al alloy, it was necessary to coextrude in the gamma phase at approximately 750 C.

Evaluation of the coextruded rods of the various alloys revealed the cladding fuel interface of the rods from the five alloys that contained approximately two w/o of either Zr, Nb, or Mo plus other additives is smooth and the cladding thickness is very uniform. On the rods of the remaining alloys there are some small variations in cladding thickness, however, the interfaces are quite irregular. The rough interfaces were undoubtedly a result of inadequate control of grain size and texture in the extrusion billets.

J. End Closure Development

Development work continued on fabrication of an unbonded TIG ^{or} ~~brazed~~ end closure. For this closure design, a V-bottomed cap is fitted into a V-notched recess in the uranium fuel. No bond is made between the uranium and the zircaloy cap, but the 90 degrees included angle of the recess design should interlock the pieces and prevent a shear failure of the cladding from the differences in thermal coefficients of expansion of the zircaloy cap and the uranium fuel. The closure is made in a single step by making a conventional tungsten-inert-gas weld with a wire preform of beryllium-zircaloy alloy between the cap and clad below the weld zone. The heat of the weld melts the beryllium-zircaloy wire preform which forms a braze joint underneath the weld and bonds to the weld. The vertical length of the braze layer between the cap and the clads is on the order of 0.100 inch. A unique feature of the joint is that the wire preform melts after the zircaloy weld has solidified and there is no beryllium present in the external weld and a true double barrier against water entry results. The technical advantages to the process are the superior quality of the pure zircaloy weld over the re-cast braze weldment on the present fuel and the fact that the two closures against water entry are truly independent. Economic advantages accrue from elimination of the braze station and the lower anticipated cost of the beryllium-zircaloy wire compared with the existing braze ring.

Development of this end closure has required the simultaneous development of a process for manufacturing beryllium-zircaloy wire. This alloy is extremely brittle and cannot be drawn by conventional wire fabrication techniques. Two processes have been successfully developed in the Battelle Northwest facilities. The wire has been made by hot swaging the cast beryllium-zircaloy clad in stainless steel at about 750 C. A second and potentially more economic process has been developed by inserting stainless steel-clad cast beryllium-zircaloy rods in multiple holes drilled in a mild steel billet and coextruding this billet at about a 9:1 reduction ratio at 800 C. The resulting coextrusion is again coextruded with a 9:1 reduction ratio. The multiple strands of beryllium-zircaloy wire are obtained by chemical dissolution of the mild steel and the stainless steel.

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Ellipticity of cladding and particularly lack of concentricity of the two claddings were problems hindering precise fitting of the end caps in the recesses in the ends of the fuels. The initial solution of this problem was to cold roll the free standing cladding after the fuel was end recessed. However, the relatively large strain in the cladding resulted in preferential hydride orientation after autoclaving at the juncture of the cladding and the uranium core. Processes have been demonstrated for obtaining a more gradual correction prior to end recessing by hot forging the fuel end.

Development of a welding process for this end closure is not complete. It has not been possible to assure flow of the braze material under the zircaloy weld with as high a degree of reliability as required for reactor service. Experimental work with different welding conditions is continuing, as well as investigation of flow aids such as vibration, post-weld heating, etc.

K. Improved Material Utilization

Tests have shown that uranium and zircaloy utilization can be increased approximately four per cent by increasing the size of the billet that is coextruded. The N-outer fuel is currently coextruded at a reduction ratio of 19-3/4:1 from 7-1/2 diameter coextrusion liner. This is being increased to a 20.4:1 reduction ratio by using an 8.1 inch diameter coextrusion liner and a larger billet. The billet length is being increased five inches. Half of the length increase is being obtained by retracting the steel follower block and half is being obtained by using a recently developed two-step coextrusion process. In the normal extrusion process, a three-inch thick graphite disc is inserted between the steel follower block and the billet assembly to clear the die cone angle of billet component material at the end of the coextrusion. In the two-step process, the graphite is not used during the first step. The coextrusion is stopped with the end of the billet still not fully coextruded and in the die cone. The die carriage is moved forward and the graphite inserted and the extrusion completed. This technique permits replacement of graphite in the first step with an equivalent increase in billet length.

Finished fuel element length is used as a means of segregating different enrichments utilized in N-Reactor. Multiple lengths are needed for each enrichment to permit reworking of fuels that are rejected during manufacture for defects near or at the end closure. The original lengths of the 0.947 per cent enriched U-235 fuels in the tube-tube geometry were 23.2, 17.4, and 11.6 inches. These have been changed to 23.2, 20.88, and 18.98 inches. Also, other appropriate lengths have been selected for each of the other enrichments for a total of nine different lengths. These changes have resulted in an improvement in materials utilization and cost savings on the order of \$77,000 per year.

L. Reduced Inside Diameter of N-Inner Fuels

Various design studies of potential fuels for N-Reactor indicate that it may be necessary to provide inner fuels with inside diameters smaller than

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the present 0.439 inch. A fabrication problem with the reduced ID is the added stress on the mandrel. With the present design ID, mandrel life is limited to an average of about five coextrusions.

Conventional tipped mandrels are not an entirely satisfactory solution for small ID's because the tip must be stationary in the die throat and overheats. The filled billet technique involves preparing a coextrusion billet with a large hole that is filled with an extrudable material. The composite billet is extruded as a rod and the filler material is removed by chemical or mechanical means. Test coextrusions have been successfully made for producing holes as small as 0.360 inch using various fillers, c.g., graphite, steel, a steel tube filled with graphite, and brass. Best success to date has been with a steel tube filled with graphite. Future plans are to use the filled billet technique with a core material that has a higher thermal coefficient of expansion than uranium, which can be dropped out as the assembly cools. Other plans are to adapt the tipped mandrel technique, but use a one-pass gas or steam cooling of the tip by drilling a small hole through the mandrel.

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