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HW-83687

This document consists of
40 pages.

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FUEL ELEMENT DEVELOPMENT COMMITTEE
ANNUAL REPORT FROM THE GENERAL ELECTRIC COMPANY

HANFORD

by:

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August 14, 1964

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GENERAL ELECTRIC COMPANY
HANFORD ATOMIC PRODUCTS OPERATION
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SECTION I

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SUMMARY OF HAPO ACTIVITIES

I. CURRENT PRODUCTION REACTORS (Excluding N-Reactor)

Over-all intrinsic fuel element quality was essentially unchanged from FY 1963. In neither year, on an over-all basis, did fuel element quality limit reactor operation or the attainment of reactor production goals. K Reactor self-supported fuels (40 per cent of total fuels produced) did not sustain any ruptures during the fiscal year. Only five failures occurred in K Reactor non-self supported fuels early in the fiscal year. The performance of bumpered fuel in the old reactors was very good (12 ruptures total). Non-bumpered fuel in the old reactors, particularly the enriched models, sustained the greatest number of failures (67 enriched and natural failures).

Emphasis during the year was in three major areas of the fuel element business for present production reactors: 1) the achievement of greater uniformity in the ALSi product coupled with programs to reduce the cost of manufacture of the ALSi I&E fuel element, 2) the development of the hot die size process as a highly attractive replacement for the ALSi process, and 3) late in the fiscal year, initial efforts to diversify the Hanford reactor products by the development of processes for producing thorium oxide fuel elements for the production of U-233.

Specific programs in ALSi I&E fuel element technology included 1) the evaluation of oil versus water-quenched uranium, which indicated that oil-quenched cores generally warped less in-reactor than water-quenched cores, 2) the evaluation (not complete) of induction heat treated cores, which when coupled with oil quenching, is reported to have economic incentives at the feed site, 3) the evaluation (not complete) of 10 inch long enriched cores which, if conversion to this length were made, would result in significant cost savings, and 4) a major effort in the design of mechanized equipment for producing, measuring, and testing self-supported fuel (40 per cent of total fuel produced) to achieve a more uniform product and to reduce labor costs of manufacture. In addition, considerable effort went into the production of high quality driver and target fuel elements for four E-N loadings in H Reactor during the fiscal year, and the production of high quality depleted fuel cores for irradiation to high exposure to produce 18 per cent Pu-240. Irradiation was initiated in July 1964.

Development of the hot die sizing process is proceeding essentially on schedule with approximately 2,000 fuel elements either irradiated or undergoing irradiation. It is believed that problems associated with diameter growth during initial irradiations will be solved by a series of irradiations currently in progress in which a variety of process parameters that are believed to effect dimensional stability now being investigated. Facilities for producing in the range of 5,000 hot die sized fuel per month (one-shift basis) will be ready for use

in the second quarter of this fiscal year and at this time, more significant quantity irradiations can be initiated. Incentives for converting the AlSi process to hot die sizing are still highly attractive with pay-outs in the range of 1 to 1.5 years on the capital investment. All of the incentives are in the area of labor and material savings over the AlSi process with no credit being claimed for better performance, higher quality, greater flexibility, or higher conversion ratio. These are added bonuses.

Late in the fiscal year, processes were developed for preparing thorium oxide target elements for U-233 production. Working closely with the feed sites to obtain a satisfactory oxide from thorium nitrate, and concurrent on-site development efforts in canning and welding of the oxide elements, resulted in the development of techniques to routinely produce thorium oxide fuel elements with a theoretical density of about 75 per cent. Six tons of thorium oxide fuel elements are currently undergoing irradiation, an additional 10 tons are currently being canned (August 1964), and recent authorizations (August 1964) will result in the canning of an additional 100 tons between now and April 1965. A production canning capacity of 35-40 tons per month will be available in October 1964.

II. N-REACTOR

N-Reactor fuel element irradiations passed from their testing phase in the K Reactor loops to their production phase in N-Reactor during the year. Test experience in the loops was extended to a total of 84 standard assemblies with a maximum tube exposure of 2100 MWD/T (peak element: 3100 MWD/T) under N-Reactor conditions. No failures were experienced in any of this fuel. In addition, 37 prototypical and developmental elements were irradiated without failure under more severe operating conditions. At the end of the year, testing in three of the four loops was concluded. A charge of KSE-5 elements was being exposed at high core temperature (570°C) in the remaining loop with a goal exposure of 2500 MWD/T.

Experience with fuel charged as first load has been generally satisfactory. Charging and some discharging of fuels has been performed without difficulty. The support system appears to be adequate. Bulk behavior of the fuel charges within the reactor confirms experimental calculated predictions with regard to flow distribution, heat generation, etc.

Performance of fuel in-reactor and in-pile test loops now allows firm planning of early initiation of higher performance operation, including exposures to 2100 MWD/T, (vs. 1300 MWD/T current goal), increased power levels (4800 MW vs. 4000 now), and introduction of high performance co-product fuel assembly.

The fuel load has operated satisfactorily. Examination of elements that have been discharged so far showed them to be in excellent condition.

One N-Reactor fuel failure has been experienced after about three months of elevated temperature test operation, with a maximum power exposure of 30 per cent of full power and negligible over-all exposure accumulation. The failure occurred through an end closure defect on an inner fuel in

the most downstream fuel in the top row of the reactor (minimum flux region). The failure was caused by a material defect in the end cap of this fuel element. The failure is not indicative of weaknesses in design or basic manufacturing processes. Probability of additional failures from this cause is low.

Continued manufacture of N-Reactor fuel elements has substantiated the validity of earlier developments in showing all materials and processes capable of continued processing with constantly improving over-all manufacturing yields. Introduction of several new models of fuel elements has been accomplished effectively. The manufacturing process and the processing equipment, which was initially designed to manufacture fuels of two enrichments, one cross-sectional configuration, and four lengths, has now been used with only very minor modification to produce fuels with four enrichments, seven lengths and several cross-sectional configurations.

Activities for N-Reactor fuel element program are centered on two objectives:

1. Developments leading to significant improvements in fuel element designs as indicated by N-Reactor capabilities for higher power, higher fuel exposure, and production of alternate products; and
2. Developments leading to improved designs and improved processes to yield consistent quality at substantial reductions in fuel fabrication costs.

While development activities required for N-Reactor fuel elements continue to be characterized by major departures from past and present development activities and technology available within the other production complex reactors, substantial use is being made of increased developments and increasing knowledge on metallic uranium technology as gained from SRP activities and the uranium technology work as sponsored by the FEDC.

As a result of progress during the year, N-Reactor Department finds itself in a firm position to take advantage of the design margins of N-Reactor and N-fuel for either higher power, longer exposure, or co-product operation. Further development along similar lines is anticipated to produce the capability for combining, to some extent, two or more of the modes of operation. Ultimately, the reactor will be capable of a degree of flexibility to produce a variety of products at low cost as a result of maximum technological exploitation of the pressure tube reactor concept.

Significant items of technical progress during the year included:

1. Technical feasibility has been demonstrated for three advanced closure processes. The three are: (a) pressure bonded; (b) self-brazed; and (c) hot-headed, projection welded.
2. Characterization of the corrosion properties of the braze alloy.

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3. Development of mathematical relationship to predict swelling behavior as a function of burnup that accounts for temperature of the core and pressure restraint by cladding and coolant.
4. Preliminary evaluation of the mechanical and thermal processing behavior, suitability of fuel element manufacturing, and irradiation behavior of the dilute iron-aluminum alloy of uranium used by the British.
5. Fabrication development and properties measurement of the standard alloy and of a number of promising alloys of uranium in preparation for irradiation testing and fuel production as part of the continuing program to develop improved and advanced fuels.
6. Demonstration of the in-reactor stability of shaped end cap in both the brazed and unbonded conditions as an improved design feature of the present closure process.
7. Development of a forged T-section fuel support with greater strength to resist charging damage, and with promise of lower fabrication cost.
8. Development of quality measurement equipment (pre- and post-irradiation), as well as improvements in the effectiveness of the over-all quality control and specification system.

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SECTION II

CURRENT PRODUCTION REACTOR FUEL

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I. FUEL PERFORMANCE

Average fuel quality during FY 1964, as measured by the relative rupture resistance of non-bumper fuel at the older reactors, improved somewhat as compared with the average level of FY 1963. Table I below summarizes some pertinent fuel performance data over the last two years on a quarterly basis.

Warp and bond count are generally considered to be the most meaningful measurement criteria which bear a relationship to non-bumper fuel performance. The rupture resistance index curve is a measure of the fuel quality based on actual rupture performance.

TABLE I

Performance Indices, Non-Bumper I&E Fuel,
B, C, D, DR, F, & H Reactors

Third Quarter 1962 = 100-Decreasing Indices Indicates Improvement				
Discharge Period	Natural Only			Enriched
	Warp Index	Bond Count Index	Rupture Resistance Index	Rupture Resistance Index
3rd Quarter - 62	100	100	100	100
4th Quarter - 62	88	74	116	119
1st Quarter - 63	70	88	83	46
2nd Quarter - 63	2	60	17	73
3rd Quarter - 63	7	60	17	8
4th Quarter - 63	76	80	50	12
1st Quarter - 64	51	102	50	23
2nd Quarter - 64	-	148	100	46

It should be noted that the performance indices listed above are for non-bumpered fuel in the old reactors. The performance of self-supported fuel in the K Reactors has been extremely good. There were no failures in self-supported fuel in either K Reactor over the fiscal year. Also the performance of bumpered fuel in the old reactors has been good. Only 12 failures occurred during the fiscal year in bumper fuel, eight of which were in D Reactor early in the fiscal year at high exposures in connection with the high Pu-240 program.

II. SPECIFIC PROGRAMS

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1. Dingot Program

At the conclusion of the dingot program, approximately 104 tons of canned dingot fuel elements and 500 tons of dingot uranium in various stages of processing remained in the feed stream. The economics favored irradiating rather than scrapping this metal. The canned dingot was charged in low-rupture potential positions in the old reactors, and the bare cores were canned by a modified process and charged in B Reactor. All of the dingot metal has been used--the last of it was charged in July 1964.

2. Oil-Quenched Uranium

The limited amount of data so far available from the bulk charging of 239 tons of oil-quenched O3N cores supports the findings of the initial production test. The oil-quenched normal uranium cores exhibited an average warp of 4.7 mils, while water-quenched cores irradiated during the same period had an average 6.1 mils warp. A fifteen-column test comparing oil- and water-quenched enriched cores has been discharged and will be evaluated soon.

3. Induction-Heat Treating

The results of a small ten-column test comparing salt-oil and induction-oil heat-treated cores showed the dimensional behavior of the two types to be nearly identical. Larger shipments of 20 tons per month of induction heat-treated cores was authorized in November, was then temporarily discontinued in March due to scheduling problems, and will be resumed in September. Eighty-six (86) tons have been charged in reactor to date, but no dimensional data is available as yet.

4. Increased Fuel Element Length

A twenty-column test to compare the dimensional stability of water-quenched six-inch and ten-inch enriched fuel elements has been partially discharged, but no dimensional data is available as yet. A similar test to compare the behavior of oil-quenched six- and ten-inch enriched cores is scheduled for reactor charging in September. A test quantity of eight- and twelve-inch normal cores, which were salt-oil and salt-water heat treated were received from NLO. The majority of these cores were mechanically straightened to remove heat-treat induced warp before machining. During canning, these cores warped excessively, indicating that mechanical straightening is not acceptable, and this material will not be reactor charged.

The feasibility of AlSi canning two cores within one jacket to produce extended-length fuel elements was demonstrated. Approximately 30 elements were tested to determine if the bond between cores could withstand the impact of discharging. At least one of these elements broke at the bond between cores indicating a weakness in this design.

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5. Depleted Uranium

Approximately 16 tons of "O" size and 16 tons of "K" self-support size depleted cores (0.22 per cent U-235) were received from NLO for irradiation to produce Pu-239 with 18 per cent Pu-240. The cores were produced by the standard fabrication route using Paducah derbies as feed material and a salt-oil heat treatment. The cores were very uniform with respect to grain size and ingot chemistry. Surface hydrogen analyses and heat treat-induced warp data were used as a basis for segregating cores at Hanford. Approximately 12.5 tons of depleted uranium were charged in B Reactor in July and 15 tons will be charged in a K Reactor loading in August. Exposures of 1200 MWD/T in B Reactor and 1900 MWD/T in the K Reactors are required to yield the necessary Pu-240 content.

6. U-233 Program

Six tons of thorium oxide target elements have been charged in the old and K Reactors to demonstrate the capability of Hanford to produce U-233 with one to five ppm U-232. The oxide was canned by two methods:

- 1) pot denitrated oxide was compacted by high-energy compaction, and
- 2) oxide produced by a modified sol-gel process was vibratory compacted.

The density of the oxide in the elements was varied between 65 and 80 per cent of theoretical to provide material for engineering studies in the reactors. The vibratory compaction of sol-gel oxide is the simplest process, requiring only low-energy vibration to achieve a uniform 75 per cent of theoretical density in the elements.

7. E-N Program

Four E-N loadings were conducted in H Reactor during FY 1964, and irradiation of a fifth and final loading was begun in July 1964. The performance of the enriched fuel in these and earlier loadings is summarized below:

E-N LOADINGS

Load No.	Period	Exposure (MWD/T)	Number of Ruptures			
			Corrosion	Mech. Damage	Other	Total
1	5/61 - 9/61	825	2	1	2	5
2	9/61 - 1/62	835	5	3	1	9
3	1/63 - 4/63	860	1	1	1	3
4	4/63 - 9/63	1083	0	0	0	0
5	9/63 - 1/64	955	0	0	0	0

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Load No.	Period	Exposure (MWD/T)	Number of Ruptures			
			Corrosion	Mech. Damage	Other	Total
6	1/64 - 4/64	676	10	0	0	10
7	4/64 - 7/64	~ 800	1	0	0	1
8	7/64 - to date					

The performance of the E-N loads has been generally satisfactory. The large number of failures in Load Number 6 are primarily attributed to the fact that this load was 100 per cent non-bumpered coupled with an unexplained decline in fuel rupture resistance. Load Number 7 was also 100 per cent non-bumpered and Load Number 8 is approximately 50 per cent bumpered. The improved rupture resistance exhibited by Load Number 7 over Load Number 6 may be due to the fact that irradiation proceeded when river water temperature was higher.

8. Uranium Specifications

Two minor revisions were made in the uranium specifications during FY 1964. The limit on iron in ingots for I&E fuel cores was increased from 185 ppm maximum to 200 ppm maximum, and the beta-heat-treat specifications were revised to provide a standard 20-second delay time from salt bath to quench. The specifications formerly required a 43-second transfer time on normal uranium and a 20-second transfer time on enriched. The revision enables NLO to heat treat both normal and enriched uranium in the Amsler-Morton furnace, thus producing a much more uniform grain structure in the enriched cores than had been observed in cores heat treated in the Nu-Sal furnace.

9. Overbore Program

Approximately 10,000 fuel elements have been irradiated in the program to demonstrate the feasibility of overboring the graphite channels to increase production. Of the 75 tons of overbore fuel irradiated to date, about 32 tons were extruded dingot uranium and the balance rolled ingot. Since September 1962, the goal exposures have been 600 MWD/T for dingot and 800 MWD/T for ingot cores. During FY 1964, five failures have been incurred in ingot overbore fuels. All were failures apparently caused by poor bonding on the ends. Examination of the elements showed the cores had larger than normal stringers which in most cases were filled with lead. Also, some compound layers were thicker than normal and were shattered. The experience to date has not been sufficient to establish statistically meaningful quality indices.

10. Projection Fuel Program

Improvements in support rail design and technology of ultrasonic welding have been made in FY 1964. The elimination of the deoxidizing of rails prior to welding and the use of oil rather than soap lubricant has improved yields and increased the strength of the welds. Prototype equipment for automatic rail feeding and lubrication has demonstrated

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that the capacity of the welders may be increased 33 per cent, and provisions are being made to install automatic rail feeders in the production plant. A new rail design is being tested that offers advantages over the present bridge-type rail of a better load-deflection curve, less flow resistance, and simplified attachment to fuel elements.

11. ALSi Process Development

The design of the present aluminum can has been revised to improve the heating rate during the ALSi dip-brazing operation. The design provides less metal to heat and a greater heat-transfer area in the base of the protective steel sleeve. Conversion to the new design is planned for all models.

Since August 30, 1963, certain categories of fuel elements, which would normally be rejected, have been categorized and charged into the first five upstream positions of the old and K Reactors. The three major upstream categories are braze-layer porosity, marred surface, and support-projection height rejects. The program has been successful, and 28,400 fuel elements have been categorized as upstream fuel and irradiated through July 1, 1964. These fuels would otherwise have been scrapped.

12. Alternate Process Development (Hot Die Sizing)

Prior investigations of processes having potentially better economics and quality than the ALSi process led to the major emphasis in alternate process development being placed on the hot die sizing process in FY 1964. The over-all economics, flexibility, ease of control, and amenability to mechanization of the hot die sizing process supported this decision.

Based on a capital investment of \$1,535,000 for a hot die size production facility to produce sufficient fuel elements for the two K Reactors on two shift, five-day week, a pay-out of the capital investment in about 1 to 1.5 years is quite realistic. This is based entirely on savings in operating and material costs over the ALSi process with no credit being claimed for better quality, fewer ruptures, higher conversion ratios, or greater flexibility. These are simply added bonuses. Capacity to produce hot die size fuel for additional reactors over and above the two K Reactors can be accommodated by increasing the number of shifts worked. Thus, by working a three shift, five-day week, hot die size fuel for five reactors can be produced with the same capital investment and about the same payout.

Complete provisional process specifications for the hot die sizing process were completed and published in document HW-77683. The specifications were developed for one model fuel for Hanford C Reactor, and slight process revisions may be required for other Hanford fuels.

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The initial production test of hot die sized fuel was charged in C Reactor on July 27, 1963. A total of 65 columns (about 2,000 fuel elements) of hot die sized test fuels has thus far been charged in C Reactor with one additional test awaiting charging and a second test being fabricated currently. Dimensional behavior of the hot die sized fuels in the initial production tests differed from that of the matched AlSi brazed control fuels. Typical of the difference in behavior are the data obtained from the post-irradiation measurements of the first in-reactor production test:

Characteristic	Average	
	HDS	AlSi
ΔOD_1 (Cap End)	11.83 mils	8.25 mils
ΔOD_2 (Midpoint)	16.63 mils	8.97 mils
ΔOD_3 (Base End)	10.33 mils	7.64 mils
$\sqrt{\text{Warp}}$	--- 2.49	$\sqrt{\text{mils}}$ ---
Average ΔID	~ 10 mils	~ 4.5 mils
Fuel Element Weight Loss	1.98 gms	1.61 gms

Radiometallurgical examination of sample fuels revealed no evidence of grain boundary tearing or bond deterioration in the irradiated fuels.

Subsequent process development work was directed toward resolution of the growth problem as well as refinement of process specifications and operating procedures. Ex-reactor tests and studies of the hot die sizing process pointed to the end bonding operation as having the greatest potential for inducing stresses in the uranium core during processing with subsequent growth during irradiation. A production test now undergoing irradiation was designed to evaluate the effect of resistance heat versus induction heat in the end bonding operation. Other production tests either undergoing irradiation or being fabricated for in-reactor testing are designed to evaluate the effects of end bonding pressure, the "skin effect" of core heat treatment, the effect of oil versus water quench heat treatment, the effect of alpha annealing of the cores, and the influence of the AlSi process on core grain structure. It is believed highly probable that one or more of these process changes will point to the solution to the growth problem by the third quarter of this fiscal year.

The end bonding process has already been converted from resistance heating to induction coil heating. Processing conditions have been established and automatically controlled equipment installed to provide a uniform, operator-independent operation in the process.

Development work was initiated on high silicon-aluminum alloy cladding. Corrosion studies of these alloys are underway. Preliminary sizing, end bonding, and welding work has proceeded satisfactorily.

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The ability to produce high-quality hot die size fuel using aluminum tubing components (40¢ per set) rather than aluminum impacts (70-80¢ per set) has also been demonstrated.

Equipment has been procured and is being installed in a pilot plant to permit the production of bulk quantities of hot die sized fuel. The major items of new equipment are an automatic nickel-plating machine, an automatic preheat furnace, a more versatile sizing press, and a component and fuels cleaning line.

SECTION III

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N-REACTOR FUELS

I. PRODUCTION STATUS AND FUEL PERFORMANCE

A. Reactor Experience - General

During the fiscal year, the first load of fuel assemblies for the N-Reactor was completed and charged in December 1963. In addition, approximately one-third of the second reactor load has been put into finished storage. The reactor achieved criticality on December 31, 1963, and had operated up to 30 per cent power by the end of the fiscal year in the course of a planned power ascension schedule. Full power operation will be achieved during this first half of FY 1965.

Experience with fuel charged as first load has been generally satisfactory. Charging and some discharging of fuels has been performed without difficulty. The support system appears to be adequate. Bulk behavior of the fuel charged within the reactor confirm experimental predictions with regard to flow distribution, heat generation, etc. Full flow loop tests combined with calculations indicate the desirability of a minor dimensional change (an increase in the inner bore of the inner tube by about 10 mils) to improve balance of water flow through the three parallel cooling paths and improve power output capability.

B. Reactor Experience - Fuel Failure

One N-Reactor fuel failure has been experienced after about three months of elevated temperature test operation, with a maximum power exposure of 30 per cent of full power and negligible over-all exposure accumulation. The failure occurred through an end closure defect on an inner fuel in the most downstream fuel in the top row of the reactor system. The failure was caused by a material defect in the end cap of this fuel element.

This fuel failure, therefore, does not detract from the adequacy of fuel design and manufacturing processes.

Review of the manufacturing and quality control processes used in the manufacture of these end caps shows the quality provisions to be marginal in preventing rejection of a small number of such defective end caps as may be produced under the present system. Sampling at receiving inspection has not caught any such defects in over one hundred thousand end caps received to date. The defect is almost certainly attributable to the usage of material too near the gross tail defect on zircaloy bar extrusions resulting in the machining of a few inner end caps which can contain some remnants of the zircaloy extrusion defect.

During the fuel manufacturing process, the defect in the end cap was covered or nearly covered at one or both ends in the brazing and welding operations. The defect, therefore, was not uncovered in the normal autoclave cycle test, and only more extended exposure to elevated

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temperature water finally corroded the thin braze layer between the bottom of the cap and the uranium core surface at the end of the fuel allowing the uranium water reaction leading to final fuel failure in the reactor.

C. Experimental Irradiation Experience

KER Loop Irradiations

Additional swelling and operational data have been obtained from two KER charges of N-fuel element assemblies. The charges were irradiated to tube average exposures of approximately 1600 and 2000 MWD/T at prototypic fuel-core temperatures and 1600 psi coolant pressure. Maximum fuel swelling observed in the 1600 MWD/T exposure test was 0.85 v/o in an inner tube (~2000 MWD/T) and 2.5 v/o in the mating outer tube (~2500 MWD/T). Maximum fuel swelling in the 2000 exposure test was 1.33 v/o in an inner tube (~2200 MWD/T) and 3.64 v/o in an adjacent outer component (~3100 MWD/T). These new data, along with previously obtained N-fuel element component and single tube element data (KSE-3's and 5's) have permitted the continuing development of an expression for fuel swelling. Analysis of the data from 130 elements has resulted in three similar expressions, one for each fuel geometry. Agreement between the calculated and experimentally determined fuel swelling is shown in Figure A in which perfect agreement is indicated by the dashed line. The expressions are:

For N-outer elements:

$$\% \Delta V \text{ in fuel} = B \left[4.15 + 550 \left(\frac{Tr}{Pr} \right) \left(\frac{1}{1 + e^{-\frac{T + 100B - 420}{20}}} \right) \right]$$

For N-inner elements:

$$\% \Delta V \text{ in fuel} = B \left[3.05 + 550 \left(\frac{Tr}{Pr} \right) \left(\frac{1}{1 + e^{-\frac{T + 100B - 480}{20}}} \right) \right]$$

And for KSE-3 and 5,

$$\% \Delta V \text{ in fuel} = B \left[3.95 + 550 \left(\frac{Tr}{Pr} \right) \left(\frac{1}{1 + e^{-\frac{T + 100B - 460}{20}}} \right) \right]$$

where B = atom per cent burnup, (1 a/o BU 8650 MWD/T)

$$Tr = \frac{\text{Vol. Mean Fuel Temp. } ^\circ K}{273}$$

Pr = Restraining pressure from outer clad and coolant, (on basis of Zr-2 strength 20,000 psi)

T = Vol. mean fuel temperature, $^{\circ}C$

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FIGURE A

COMPARISON OF MEASURED AND
CALCULATED FUEL SWELLING

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A total of 84 standard N-fuel assemblies has now been irradiated in the KER loops under N-Reactor conditions up to 2100 MWD/T average tube exposure. Maximum exposure for a single element is 3100 MWD/T. No failures were experienced in any of this fuel. This concludes the irradiation testing of standard fuel in the KER loops now that N-Reactor is fully loaded and operating. At the end of the year, only special fuel elements were being irradiated in the loops with only one loop scheduled to continue operating into FY 1965. Figure B summarizes all KER loop irradiations to date.

D. Other Experimental Irradiation Experience

1. Fluted Single Tube Irradiation

A fluted outer surface on tubular elements as a means for extending the total exposure capability of metallic uranium fuel has been examined in irradiation tests in the M-3 high pressure loop facility of the ETR. Two elements of N-single tube size containing eight flutes about 0.1 inch high generated by circular arcs were exposed. The elements operated at a specific power of about 140 kw/ft. with a corresponding metal temperature of 550 C. After an exposure of about 2500 MWD/T, one of the elements failed and the test was terminated. Interim weight measurements after each reactor cycle permitted determination of fuel volume change as a function of exposure. The elements had undergone approximately 2.4 v/o increase as a result of this exposure. No definitive information has been obtained from the failure as yet. The successful in-reactor irradiation of a heavier element with higher enrichment (see Section III-G) as needed for co-product operation has obviated the need for the so-called shaped or fluted single tube element as had been considered necessary earlier.

2. Zr-2 Cladding Strain Study

Localized thinning or necking and ultimate failure of Zr-2 cladding has occurred on two fuel elements irradiated in pressurized water, and on several rods irradiated in NaK-filled capsules. Irradiation of 91 zircaloy-2 clad uranium rods, 1/2-inch diameter, with intentional striations in the inside surface of the cladding was conducted to study the effects of cladding thickness, cladding thickness variation, and cladding temperature on susceptibility to failure. The results indicated that at temperatures below 350 C, the total cladding strain limit before necking failure is only 1.5 per cent. Above 350 C, the strain limit increases rapidly. The observed low strain limit is probably related to irradiation damage, and the increasing capability for strain with increase in temperature is believed due to annealing of irradiation damage. At all temperatures, the presence of single striations greater than 10 per cent of the cladding thickness reduced the limit of the total cladding strain to one per cent. At greater striation depths, the strain limit is further reduced.

3. Irradiation Testing of Metallic Uranium with Submicron UC Dispersion

Three NaK-filled capsules are being irradiated in the ETR to determine irradiation performance of fuel rods produced from chill cast

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Figure B SUMMARY OF H-FUEL TEST IRRADIATIONS

uranium shot relative to fuel rods produced from slowly cooled arc melted uranium. The rods produced from shot have a dispersion of 0.5 microns or less UC, while the normal UC size is 2-5 microns. One capsule completed exposure to 0.3 a/o burnup at a maximum temperature of 550 C and is awaiting examination. The other two capsules are operating at 600-625 C.

E. Production Status

Continued manufacture of N-Reactor fuel elements has substantiated the validity of earlier developments in showing all materials and processes capable of continued processing with constantly improving over-all manufacturing yields. Introduction of several new models of fuel elements has been accomplished effectively. The manufacturing process and the processing equipment, which was initially designed to manufacture fuels of two enrichments, one cross-sectional configuration, and four lengths, has now been used with only very minor modification to produce fuels with four enrichments, seven lengths, and several cross-sectional configurations.

The initial quantity of fuels consisted of 372 tons of 0.947 per cent U-235 enrichment. An additional 75 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 was started into the process to provide additional reactivity needed to achieve full power on the planned schedules. Also, 30 tons of 1.25 per cent U-235 enrichment uranium was started into fabrication for conversion to co-product test elements (minimum of 36 tubes) to demonstrate N-Reactor capabilities for co-product operation. Charging of all of these higher enrichment elements is scheduled to be accomplished during August 1964.

Conversion costs for the fuel manufacturing process have continued to show the predicted substantial improvements as shown on Table A. These gains in unit conversion costs are the result of improved manufacturing yields both in terms of total material utilization and piece yields as tabulated in Table B. Further contributions to reduce unit costs have been obtained through significant reductions in the cost of the zircaloy components. A 30 per cent reduction in average price of zircaloy components was gained during the year compared with fiscal year 1963, for a total saving in excess of \$500,000.

F. Manufacturing Process Highlights

The process for fabrication for N-Reactor fuel elements has remained essentially constant over the past year. A pictorial flow sheet is shown in Figure C and a picture of a finished fuel element showing the auto-claved inner and outer 24-inch long fuel tubes just prior to final assembly is shown on Figure D.

The problems with fabrication of zircaloy supports with inner and outer fuel elements as reported last year have been completely overcome. While the major portion of this hardware was produced on site with ductile

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TABLE A

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 - Actual	\$ 7.88/lb. for 342 tons
Target	8.18/lb. for 334 tons
FY 1965 Targets	\$ 5.95/lb. for 512 tons
	6.43/lb. for 420 tons

TABLE B

SUMMARY OF PIECE AND WEIGHT YIELDS

<u>Major Losses</u> <u>(Piece Yields)</u>	<u>FY 1964</u>			<u>First Load</u>		
	<u>Outer</u>	<u>Inner</u>	<u>Total</u>	<u>Outer</u>	<u>Inner</u>	<u>Total</u>
Coextrusion Bond	95.2%	98.6%	96.9%	96.1%	92.1%	96.6%
Extrusion Surface	96.6	99.5	98.0	99.8	99.9	99.8
Warp	99.8	94.3	97.0	99.8	97.3	98.5
Clad Thickness	88.1	97.4	92.8	87.0	99.0	93.0
End Closure (Partially Recoverable)	89.5	92.3	90.9	87.0	92.7	89.9
Shop (Weight) Yield	82.6	76.5	80.6	51.9	66.7	56.5
Uranium (Weight) Utilization	67.2	65.1	66.5	42.0	56.7	46.6

CO-EXTRUDED FUEL

REL FLOW SHEET

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FIGURE C

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FIGURE D - N-FUEL ELEMENT

zircaloy strip fabricated by Hanford Laboratories, one commercial vendor is now capable of fabricating ductile strip and some of the components are being procured off site. Strength of the outer supports has been increased through use of 50 mil thick zircaloy strip as compared with the 40 mil thick earlier design. Feasibility of forming this heavier material is further evidence that sufficient technology of making ductile zircaloy strip has been acquired. The use of this stronger support as now used on the second reactor load fabrication assures complete compatibility of fuel with reactor charging system.

II. DEVELOPMENT PROGRAM AND PROCESS TECHNOLOGY

A. General

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

1. Developing the technology and fuel fabrication processes for the co-product mode of operation of the N-Reactor, i.e., a more highly enriched outer uranium driver fuel element containing a center rod of zircaloy and aluminum-clad lithium containing target material for tritium production.
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 through improved 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 use of process materials; (c) reducing costs of purchased materials through improved specifications, definitions of quality, development of new sources.

Knowledge of fuel characteristics obtained to date assure adequacy of present design at increased exposure (2100 MWD/T) for the second loading versus 1300 MWD/T for the first load, and operation at increased power levels (4800 MW versus 4000) within two to three years.

B. Uranium Billet Fabrication

The uranium billet core is presently manufactured by casting an 11-inch diameter by 30-inch long ingot containing a nominal 150 ppm of iron and a nominal 100 ppm of silicon, triple-beta heat treating the ingot, and primary extruding to billet blank stock. The coextrusion billets are machined from these primary extrusions and coextruded with no further heat treatments. The major billet-related problem in the present coextrusion process has been zircaloy-2 cladding thickness variation that is caused by local variation in resistance to deformation of the uranium, resulting to a large part from the residual ingot structure. The bulk of the cladding related rejects listed in Table B are from this cause.

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A billet fabrication process based on primary extrusion of larger outside and inside diameter ingots over tip mandrels is being developed. The ingot size selected is 13-inch OD by 6-1/2 inch ID for primary extrusion to a 7.2 inch OD by 2.6 inch ID. The thin wall ingot design results in more effective heat treatment before primary extrusion and the extrusion over the tip mandrel results in more alpha working of the ingot structure, particularly on the ID surface. The purpose of this program is to more effectively obliterate the original uranium grain domains in the cast ingots that are believed to be the principal cause of clad thickness variation, particularly in the ID of the N-outer tube. Initial coextrusion results from billets fabricated by this process are favorable. Maximum cladding thickness variation measured circumferentially on the fuels was reduced from 0.0073 inch for billets fabricated by the normal route to 0.0051 inch for billets fabricated by the tip mandrel route. Some difficulty was encountered with warp of the primary extrusion which resulted in a finished billet yield of about 20 per cent of lack of cleanup and wall thickness variation after billet machining. Other primary extrusion tests are planned to eliminate this difficulty, which is believed solvable by minor modifications in primary extrusion tooling.

An additional technique for increasing the alpha working on the ID surface of the uranium ingot is through use of a cupping technique. An 11-inch diameter ingot is used with a very small pilot hole of about one inch diameter. This pilot is hot-expanded to 2.7 inches and the ingot extruded. No primary extrusion difficulties have been encountered using this process and initial coextrusion results indicate that some gain in cladding thickness variation control may be obtainable using this process.

A third billet fabrication process for overcoming clad thickness variation is to install a highly worked uranium sleeve in the ID of the billet. This highly worked sleeve should have essentially no hard and soft areas from ingot memory and should act as a barrier to any uneven deformation tendencies in the large outer sleeve. The concept has been successfully tested by double extrusion of the uranium stock to fabricate the inner sleeve and is now being tested through a single primary extrusion by what should be a reasonable production route.

Uranium Fabrication Development Programs

Tabulated below is a list of the year's significant activities in the N-Reactor uranium fabrication development programs initiated by Hanford and/or the Feed Materials Production Center.

<u>Program</u>	<u>Status</u>
<u>Primary Extrusion Conditions</u>	
Effects of ram speed.	Aimed at improved ID working; studies at 18 and 30 inches per minute--did not show significant clad variation improvement.

Program

Status

Primary Extrusion Conditions (Cont'd)

Effects of decreased
mandrel temperature.

Decreasing mandrel temp. by approxi-
mately 300 F did not significantly
change fuel quality. Minor improve-
ment was obtained when combined with
high ram speed condition.

Increase of included die
cone angle from 120° to
140°.

Aimed at more severe metal working;
this test did not result in significant
changes in fuel quality.

Process Route Variations

Ingots - solid cast and
drilled.

By drilling solid ingots the last and
most void-prone metal is removed,
casting costs are decreased and ingot
concentricity is improved. This is
present process route.

Uranium shot.

Intended to improve second-phase
particle dispersion and reduce uranium
process costs--shot ordered and received
and preliminary test work planned.

Sleeved billets.

By making two component billets,
greater reduction ratios with resultant
grain refinement is possible, thus
leading to improved cladding thickness
variation--preliminary tests have been
good and a production test is now
underway.

Rapid ID quench.

Forced water quench of ID expected to
improve uniformity of refinement by
breaking up surface adherent steam--
preliminary results were not promising
and program discontinued.

ID cupping of NIT
to NOT ingots.

Preliminary work has progressed through
coextrusion evaluated and shows that
the increased work performed by forcing
a tip through a small ID ingot may
result in significantly improved con-
trol of cladding variation.

Primary extrusion of 13"
OD ingot.

No significant improvement resulted from
the increased reduction ratio probably
because of the large ingots' poorer heat
treat characteristics.

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<u>Program</u>	<u>Status</u>
<u>Process Route Variations (Cont'd)</u>	
Primary extrusion of 13" OD x 6-1/2" ID ingots over a tipped mandrel.	The superior heat treat characteristics of this geometry combined with the shrinking of the ID over a tipped mandrel have resulted in improved cladding quality in initial coextrusion testing. A larger production test is being planned.
<u>Metal Improvement</u>	
Silicon additives.	Test coextrusions of silicon levels of approximately 150, 250, and 420 ppm. Some indication of improvement in control of cladding thickness variation was found.
Physical testing.	Program started with objective of detecting and evaluating metal variations before coextrusion. This work currently directed toward hardness traverses of the inner bore of NOT billets.
Stringer rating.	A system of stringer ratings in coextruded uranium has been underway for approximately five months in an attempt to detect variation in metal quality.
Arc melting.	Billets from arc melted ingots have been made in an effort to improve second phase dispersion. Coextrusion results are not yet available.
<u>Cost Savings Programs</u>	
Reconditioning of billet surfaces.	Coextrusions of billets with mechanically removed surface defects indicate that such billets are suitable for NPR use and specification allowances are being made.
Coextrusion of short billets.	Cost analyses have been made showing that savings are available and two categories of short billets are now acceptable.

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<u>Program</u>	<u>Status</u>
<u>Cost Savings Programs (Cont'd)</u>	
Hydrogen sample.	Billet yields have been improved by revising the required billet hydrogen sampling. Sample location was moved to the first full cross section at the rear of the extrusion.
Evaluation of high-phosphorous concentration in uranium.	An inadvertent introduction of phosphorous into the feed stream initiated a fabrication and irradiation study which resulted in the use of most of this material.
Evaluation of low carbon billets.	A casting problem in the feed stream resulted in a large number of sub-standard billets. Investigation of fuel fabrication risks resulted in the ultimate use of an appreciable amount of the material.

Uranium Billet Specification

The uranium billet specification systems are being improved in two significant ways. The entire format of the specifications is being changed. The existing specifications are a combination of product and process requirements that are issued by Hanford and detail some, but not all, of these parameters. Under the new format, Hanford will essentially describe the final and, where necessary, the intermediate products needed to fabricate finished billets. These descriptions will be primarily drawing based. The supplier (The Feed Materials Production Center) is preparing and will issue the necessary process specifications for the control of the intermediate products that affect final billet quality. The combined Hanford and FMPC documents will encompass all known critical product and process requirements for fabricating uranium metal billets. When physical tests for adequately characterizing the uranium are developed, these will be added to this specification system.

The other change in specification is the establishment of product quality limits that encompass the output of the FMPC process whenever that process is in an optimum state of control and whenever the extremes of the distribution are acceptable input for N-fuel coextrusion. Additional gains from developing specifications of this type are the control of the central tendency of the distributions, i.e., greater uniformity, reduction in the number of waivers and associated paperwork when the product is being waived is really no different from that normally produced, and reduced billet cost. The approach selected for implementing this program is the use of a two-limit system where the material between the first and second limit represents the fringe of the normal distribution. These billets will be identified to enable

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observation of fabrication performance. Class II limits have already been established for accepting billets up to 1/8 inch short. Consideration is being given to accepting billets as much as two inches short. The cost of scrapping or replacing the occasional short billets that are produced is about \$75 to \$110 per billet, depending upon the geometry. This cost is about equal to the extra cost of coextruding billets that are two inches short. The additional coextrusion costs are caused by lower material utilization in the coextrusion billets because of a constant length of the end defects. Plans are to extend this two-limit specification system to billet surface defects, other billet dimensions and some processing specifications.

Changes in NPR Uranium Billet Specifications-July 1, 1963 Through September 30, 1963

<u>Date Change Effective</u>	<u>Specific Change in Limits</u>	<u>Reasons for Change</u>
7/63	Briquett allowance in melt charges increased from 15 to 25 per cent.	Amount of chips has increased from ingot machining to improve concentricity. Melts for NPR ingots should utilize an amount of chips equivalent to the amount generated in producing NPR billets.
	OD machining of ingots w/o wall variations of less than 0.080" accepted if OD machined to 10.825" $\pm$ 0.025" and ingot extruded from special 11.000" liner and ingot top end radiused or chamfered.	Process permits salvage of eccentric ingots. The ingot to liner clearance was reduced to improve extrusion concentricity and, therefore, billet symmetry.
	Isotopic content and tolerances for 1.25 enriched uranium added to specification.	Included in anticipation of a production order.
	Drilling of solid ingots required before beta heat treatment.	Coextrusions have shown that heat treating ingots before drilling results in a lack of structure refinement and excessive cladding thickness variation at the ID.
	Extrusion cutting schedule changed for both NO and NI to eliminate length of rear end crop and to permit cropping of a full section with no length requirement.	Changed to permit utilization of all fully worked uranium.

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<u>Date Change Effective</u>	<u>Specific Change in Limits</u>	<u>Reasons for Change</u>
8/63	Hydrogen limit of 2.5 adopted if the sample was obtained from the rear end defect of the primary extrusion to permit cutting of full length billets.	The change was based on comparative data that showed the hydrogen content of the rear end defect was about 0.5 ppm higher than the hydrogen content of the billets.
9/63	Chemical composition for the mold coating removed.	Selection of mold coating materials left to the discretion of the supplier provided HAP0 is notified of any proposed changes.
	Phosphorus impurity limit of 35 ppm established.	Lowest limit compatible with suppliers' capabilities adopted to minimize irradiation performance risk.
1/64	Melting and casting power-time cycle changed to permit up to five minutes hold time after power-off before pour.	Hold time increased from three minutes to provide additional time for liquation of unmelted impurities.
2/64	Dimensions of billets for 1.25% U-235 enriched coproduct driver tubes added to specification.	A special billet was designed to provide flexibility for fabricating either coproduct driver fuel elements using a uranium sleeve concept or N-outer geometry spike enrichment fuels.
4/64	A Class II limit of 75 ppm for phosphorus for 10 per cent of the billets added to the chemical specification.	Fabrication and irradiation tests at Hanford showed no detrimental properties at 75 ppm. Double limit of 75 ppm and 35 ppm used to insure uniform detection of phosphorus containing uranium feeds.
4/64	A Class II billet length 1/8 inch less than standard adopted for 10% of the billets.	The occasional short billets produced can be used and the cost of remelting and replacing the billets saved.
6/64	Carbon specifications changed from 350-650 ppm range to 365-735 ppm.	Carbon specifications changed to encompass normal distribution encountered using specified melting and casting cycles.

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Date Change Effective	Specific Change in Limits
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6/64	Specifications for 1.95 U-235 enriched uranium billets for coproduct driver tubes issued	The billet dimensions, enrichment and alloy composition for these billets are different from standard billets.
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C. Physical Tests of Uranium Properties

Very little work has been done to characterize the physical properties of the N-Reactor uranium billet stock, particularly the hot physical properties. Programs initiated now include measurement of hot hardness and metallographic examination of uranium stock after coextrusion for stringer content. Considerations of additional tests such as impact tensile, ball creep, etc., are being considered. The purpose of these programs is to develop a basis for fully characterizing the uranium and to permit development of true product specifications for uranium billets.

D. Alternate Uranium Composition

Major emphasis this year in the continuing program of controlled irradiations of alternate fuel alloys was on the so-called "British" alloy. Swelling resistance was compared up to 1530 MWD/T for fuel elements fabricated from standard composition and from the "British" alloy containing nominally 350 ppm Fe and 800 ppm Al. Four elements of each composition were irradiated in a KER loop at 570 C core temperature. Pre-irradiation measurements included structural characterization of the ingots, primary extrusions, coextruded fuel and completed fuel elements. The increased iron and aluminum content caused the "British" alloy to have a higher recrystallization temperature, a smaller as-extruded grain size, a higher hot hardness at 300 to 400 C, and a higher number of second phase compound inclusions. No significant differences were observed in the beta heat treated grain size or the clad-to-core bond appearance.

<u>Element Number</u>	<u>% Volume Increase</u>	
	<u>Stand U</u>	<u>"British" Alloy</u>
1 (Upstream)	0.42	
2		0.86
3	1.40	
4		1.18
5	2.12	
6		1.15
7	1.95	
8 (Downstream)		1.05

In general, the "British" alloy elements swelled only about 60 per cent as much as the immediately adjacent standard elements (excluding the anomolous upstream standard piece). These results argue well for

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future plans that require fuel to operate at much higher temperature or longer exposure. A second charge identical to the one described above is currently being irradiated to a goal exposure of 2500 MWD/T in another test loop. During FY 1965, several charges of "British" alloy fuel elements will be irradiated in N-Reactor.

Alloy Development

In addition to this relatively large-scale work, work was almost completed on the preparation in the Laboratories of fourteen different alloys for NaK-capsule irradiations. The relative swelling is to be compared as a function of composition, heat treatment, and irradiation conditions.

Eight of these alloys were received from the Mallinckrodt Chemical Works. These alloys, produced as dingots from the bomb reduction process, were fabricated by gamma extrusion of the dingots into two-inch diameter rods. To provide small coextrusion billets, the rods were cut into six-inch lengths. Prior to final machining, the billet blanks of the U + Si, U + Al + Si, U + Al, and U + P alloys were beta heat treated and water quenched. Cracking occurred along the radially-oriented grain structures in the first three alloys; the U + P alloy did not crack. Quenching into salt or air cooling to lower the cooling rate still resulted in cracking of the billets. The cracked billets and original stock of the U + Si + Al and the U + Al alloys are being remelted and cast in an effort to produce sound coextrusion billets.

The beta heat treated U + P and the remaining alloys with Zr or Nb additions and three sound U + Si billets, all in the as-gamma extruded structure, were coextruded with Zr-2 cladding to 0.6 inch diameter rods. Cladding thickness variation was good on all the rods except the U + P, U + Si, and the U + Nb + Al alloys. From previous experience with the irradiation instability of non-uniform Zr-2 cladding, these two alloys will be reclad before they are irradiated.

The coextrusion of the U + Nb + Al alloys was difficult. Using coextrusion preheat temperatures in the high alpha range, the finished rod had a longitudinal cladding thickness variation ranging from 0.020 to 0.040 inch. The variation is believed to be related to the greater "stiffness" of the U + Nb + Al alloy compared to the Zr-2 in the alpha-phase temperature of the uranium. Attempts to alpha coextrude the alloy using a "stiffer" alloy of Zr + 3Nb + 1Sn for the cladding were unsuccessful. The problem was solved by coextruding with Zr-2 cladding in the low gamma-uranium temperature range (750 C).

Six other alloys of uranium with varying amounts of Fe, Si, Al, and C are now being produced using either vacuum induction or arc melting facilities at Hanford.

The co-product mode of operation for N-Reactor will require an outer uranium driver fuel element with a uranium-235 enrichment of about

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two per cent and a uranium fuel wall thickness of 0.48 inch compared to a uranium wall thickness of 0.30 inch on the current N-outer fuel element. The estimated exposure conditions for these driver fuels are central metal temperatures of 565 C (600 C peak) and an exposure of 1500 MWD (2200 MWD maximum). Performance of the present uranium alloy for this type of service would be marginal; therefore, studies have been initiated to determine the effects of altered fuel compositions upon fuel element fabrication, corrosion behavior, and irradiation swelling resistance. Prototypical test fuels will be fabricated for irradiation using the following four compositions of iron, aluminum, and silicon additives:

<u>Iron</u>	<u>Aluminum</u>	<u>Silicon</u>
350	800	---
250	250	250
250	---	350
150	---	360

These alloys have been selected as most promising based on alloy development results obtained by SRP.

The billet fabrication routes that will be tested as part of this program include:

1. 11-inch ingot, beta heat treated, extruded, machined;
2. 11-inch ingot, extruded, machined;
3. 11-inch ingot, beta heat treated, extruded into the uranium sleeves for a two-part billet assembly;
4. billets cast directly to size, heat treated.

E. End Closure Development

The present production end closure for N-fuels consists of zircaloy-2 ring brazed into the recessed end of the fuel element using a five per cent beryllium and zircaloy-2 alloy as the brazing material. The irradiation tests as well as ex-reactor tests have proven this closure design to be sound and reliable. The problems with this closure design have been as follows:

1. The high cost of the material consumed. The cost of the four braze caps and rings per fuel assembly averaged about \$25.00 (or 27 per cent of material costs) during the year.
2. The brazing process uses induction heating in a vacuum controlled by optical temperature measurements and is difficult to control. As a result, yields have not been high because of shrinkage porosity in the cast braze, non-uniform flow of the braze material, and the washing of uranium contamination into the end closure because of the uneven braze flow.

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Programs for improving the cost of materials include investigation of powder metallurgy braze rings with a cost savings potentially on the order of 25 per cent for braze rings, and the use of simplified cap designs which have a cost savings potential of about 20 per cent in cap costs. Process yields are being improved by optimizing the dimensions of the components and the process parameters in order to promote more uniform braze metal flow. Over half of the end closure rejects listed in Table B were caused by small casting voids near the top of the braze. These small isolated voids can cause arc interruptions during welding, but rework processes have been developed to recover the majority of these fuels.

F. Advanced End Closures

Development of three advanced closure processes was essentially completed. The three processes have been taken to a point where technical feasibility has been demonstrated.

The gas pressure bonded end closure was developed to permit closure in the alpha temperature range of uranium. The process is capable of producing a closure which, with the weld removed, will withstand a 750 F, 72-hour autoclave test. Metallographic examination of the bonding has shown it to be complete and free of voids.

Fuel elements with closures fabricated by the "self-brazing" process, both before and after beta heat treatment, were successfully irradiated. Additional testing of the closure involved exposing the braze layer by machining the end of an element and then autoclaving it for 72 hours in 570 F steam, with no apparent attack. The same element was transferred to a high-temperature recirculating loop where it was subjected to 108 days in 530 F water at pH 10. The braze layer appeared unaffected at the end of that period.

The hot-headed, projection welded closure process was improved by minor changes in welding fixtures and by studies to determine the amount of copper on the end caps that will provide the most reproducible metallurgically sound joint. Work to date indicates that the process is feasible for both inner and outer fuel elements.

G. Shaped End Closure

The V-bottom modification of the current flat end cap was irradiation tested in the KER loops. In this design, the V-bottom cap fits into a V-shaped recess in the uranium. Strain from differential thermal expansion of the zircaloy end cap and the uranium are accommodated over about a 0.25 inch transition region, thus making the closure less susceptible to shear-type failure.

Thirteen (13) co-product driver test elements with this closure in the fully brazed condition were successfully irradiated to an average exposure of 1350 MWD/T with a maximum uranium core temperature of 530 C. Fuel volume increases ranged from 0.25 to 3.3 per cent. Visual

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examination showed the closures to be in excellent condition. The same closure has been used in fabricating a 38-tube charge of the same type of fuel for irradiation in N-Reactor beginning in August 1964.

In addition, several fuel element inners were fabricated with the V-bottom closure in the unbonded condition. These were irradiated in two other loops to a maximum of 1900 MWD/T with no evidence of abnormal behavior. Earlier irradiation of some unbonded flat end cap elements resulted in failure by shear of the cladding at the cap-to-core junction. Future work on unbonded end caps is being held up pending evaluation of a semi-brazed design now under development.

H. Corrosion of Braze Materials

Corrosion studies are being continued on zirconium-base alloys containing varying amounts of beryllium and uranium in 360 C water at pH 7 and pH 10. Several alloy systems have been added to the study: Zirconium-beryllium with iron; copper and thorium; zirconium-iron-nickel; and zirconium-iron-chromium and nickel. This work is aimed at defining the relation of braze metal corrosion rates versus composition for the current braze metal and other braze metal compositions.

Because a large mass of data have been accumulated, a FORTRAN program was written to analyze the results. This program, called CORAN (for corrosion analysis), calculates the surface area, weight differences, weight gain/loss per unit area, and normalizes the light data.⁽¹⁾ Normalized light data plotted as a function of time yield a curve very similar to the weight gain curve up to the point of breakaway corrosion. These light data are analogous to the optical properties of pure metals and represent the approach to the stoichiometric condition. The basic program is being enlarged to incorporate a regression analysis of the optical properties as a function of weight for each measured wave length. This analysis will show any deviation from ideal and should indicate these alloys which have either improved or poorer corrosion properties.

To date, the zirconium alloys with varying beryllium content have amassed 3400 hours in 360 C pH 7 water. After 2900 hours, the 1.13 per cent Be alloy began to rapidly lose weight the same as in the pH 10 test after 750 hours. Zircaloy with 0.5 per cent uranium had the same weight gain after 3500 hours as zircaloy-2 after 3400 hours. Additions of iron to a Zr-2 + 4.5 per cent Be alloy shows decrease in the weight gain with increasing iron content, up to 12 per cent Fe. The Ni-Cr-Zr-2 alloys had half the weight gain of Zr-Cr-Fe and Zr-Cr-Fe-Ni alloys. Copper additions up to 14 per cent Cu resulted in large weight gains for the Zr-2 plus 4-1/2 per cent Be alloy. Thorium additions of only 120 ppm resulted in large weight gains for the basic zircaloy-2, 4-1/2 per cent Be alloy.

(1) The measure of an optical property of the specimen surface which is analogous to absorption and/or reflectance.

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I. Coextrusion Technology

Incomplete bonding of the zircaloy to the uranium still occurs periodically although very significant reductions in this reject rate have been obtained. Much of the problem is due to rupture of the weld in the copper can enclosing the billet assembly. Use of a heavier copper front-end plate has been responsible for much of the gain in bonding yield obtained. Further gains will be obtained by redesigning the copper weld joint. However, unbonding is a more complex problem than just rupture of the copper closure, and fundamental studies on the bond and the bonding mechanisms will be needed for complete solution of these problems to be obtained.

Other coextrusion efforts are being concentrated on improving the lubrication of the billet assembly. The bulk of the extrusion defects listed in Table B is caused by dimpling of the inner cladding. Changes in the lubricant, lube application techniques, and preparation of the copper can surfaces have reduced the yield losses from dimpling from a high of 10 per cent to less than one per cent.

J. Forged Fuel Support

Development of stronger and less expensive outer fuel tube supports, fabricated by means of hot forging techniques, was conducted during the year. One type of outer fuel tube support is hot forged directly from 1/8-inch diameter zircaloy-2 wire to finished dimensions. The increased depth of section (Tee shape) in the crown area markedly increases the support strength and permanent set characteristics.

Supports formed thus far have been forged on a 600 KVA resistance welder because of easily and accurately controlled heating conditions coupled with up to 18 tons forming force. The 1/8-inch diameter zircaloy-2 wire is resistance heated into the beta phase (950 C estimated from color) and formed by a 15 ton blow. Total heating time is 1.5 seconds followed immediately by the forming blow. The part experiences an extremely rapid quench in the die. Approximately two seconds elapse after the die closes till the part is cool enough to be held by hand. For surface finishing, the part is given a light sand blast and bright etch.

Approximately 300 pieces have been autoclaved in 400 C steam for 72 hours with no evidence of abnormal corrosion behavior. Ten pieces are in an extended autoclave run in 360 C water, and had accumulated 504 hours at the last examination. No evidence of abnormal corrosion behavior was seen.

K. Nondestructive Test Equipment

1. Billet Enrichment Testing

Testing of all N-fuels billets has been instituted by spectral analysis of the normal gamma radiation. The test will differentiate with a high degree of confidence between the normal

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enrichments processed into N-fuel (natural, .947, 1.25, and 1.96). In the test, a scintillation detector is introduced in the bore of an end fuels billet. A one-minute count is made on the 185 KEV peak emitted by the U-235 enrichment.

2. X-Ray Fluorescence Testing

A new test for detecting and classifying uranium contamination in the end closures of N-initial fuels is being used on a developmental basis at Hanford. The end closure is scanned with a collimated x-ray beam from a 150 kv x-ray source. The x-rays excite any uranium atoms in the field, resulting in the emission of the characteristic electromagnetic energy of the K-series electrons in the uranium atom. This characteristic electromagnetic energy is sensed by a sodium iodide crystal sensor, sent to a single channel analyzer, and displayed on a recording chart. The closure is rotated under the x-ray beam to scan 100 per cent of the inner and outer weld beads. Separate sensors and signal channels are used to show the distribution of uranium in inner and outer beads. The tester has a threshold of detection between .5 and one per cent uranium. Production capability of the tester, in the prototype stages, is in the range of 20 to 30 pieces per hour. It is probable that x-ray fluorescence testing will supplant the present methods for detection and classification of uranium in the closures, which are autoradiography for detection and alpha counting for classification. The prototype equipment and a schematic diagram of its principle are shown in Figures E and F.

3. Post-Irradiation Fuel Examination Facility

A project was authorized to provide special handling and post-irradiation measurements equipment for irradiated 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 will be located in the N-Reactor storage area using the water 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 now used in the facility associated with the KER test loops will be upgraded and used in the new facility.

L. N-Fuels Quality Costs

The costs of assuring a quality product are being collected by the N-RD Fuels organization. These costs fall into three basic categories:

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FIGURE E - X-RAY FLUORESCENCE-TEST EQUIPMENT

SECONDARY RADIATION ENERGY SPECTRUM

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FIGURE F - X-RAY FLUORESCENCE-SCHEMATIC

1. Costs of failure - those costs incurred due to failure of the product to meet specifications as fabricated (internal failure), or costs of failure of the product in-reactor (external failure).
2. Cost of appraisal - the costs of tests and inspections to assure that all product shipped is of the required product quality.
3. Costs of prevention - the costs incurred by efforts to prevent failure costs. These include engineering effort and improvement of process capabilities, management of the quality control function, and design, implementation and maintenance of the quality system.

The costs of quality are examined with the philosophy of alternate investment. For instance, investment in prevention costs should yield a return in measurable reduction in scrap and rework, which are internal failure costs. Similarly, investment in appraisal costs should yield a return and reduction of reactor problems, which are external failure costs. Any investment in one area of quality cost must be balanced by a greater return in another area of quality costs, so that the total quality costs can be reduced. At present, quality costs constitute about 30 per cent of the conversion cost of the N-fuel.

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