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REPORT TO THE WORKING COMMITTEE OF THE
FUEL ELEMENT DEVELOPMENT COMMITTEE FROM
THE GENERAL ELECTRIC COMPANY - HANFORD

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REPORT TO THE WORKING COMMITTEE OF THE
FUEL ELEMENT DEVELOPMENT COMMITTEE FROM
THE GENERAL ELECTRIC COMPANY - HANFORD

by:

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

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CURRENT REACTOR PRODUCTION FUELS

Uranium Core Production Data

Quality of Uranium Cores

Visual Inspection of Virgin Cores

Visual reject rates for the period September 1 through November 30, 1964, were 0.06 per cent and 0.08 per cent for normal and enriched cores, respectively. This is approximately the same as for the last reporting period.

UT-2 Rejects

The UT-2 grain size rejects for the same period were 0.01 per cent and 0.03 per cent for normal and enriched cores, respectively. The surface tester rejects were 0.03 and 0.02 per cent for normal and enriched cores, respectively.

Grain Size

The average grain size of cores received during the same period was between A-6 and A-7 for both normal and enriched uranium.

Uranium Specifications

As mentioned in the last report, a sample exchange between HAPD-IPD, Hanford Laboratories, NLO, and New Brunswick Laboratories was conducted to resolve a difference between NLO and IPD results on magnesium analyses. The results of the exchange are:

Sample Exchange for Mg Analyses

<u>Sample</u>	<u>NLO</u>	<u>IPD</u>	<u>HLO</u>	<u>NBL</u>
1	0.5 ppm	~ 5.0 ppm	~ 5.0 ppm	1.7 ppm
2	2.0 ppm	~ 0.5 ppm	2.0 ppm	1.7 ppm
3	4.0 ppm	~ 0.2 ppm	2.0 ppm	1.2 ppm
4	2.0 ppm	< 0.2 ppm	1.0 ppm	1.8 ppm

The results indicate the NLO analyses are probably better than the IPD results, and that the current uranium will not meet the maximum limit of one ppm magnesium as requested by the Chemical Processing Department at HAPD. CPD is developing a process to separate magnesium from strontium 90, and, assuming this is successful, the limit on magnesium may be established at ten ppm rather than one ppm. The limits on calcium, strontium, magnesium, cesium, and cerium mentioned in the last report have not yet been included in the uranium feed specifications.

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Induction Heat Treating

The charging of twenty tons per month of induction heat treated O3N cores was resumed in November. Post-irradiation data on the first 86 tons charged is not yet available.

Fuel Performance

Seventeen normal and ten enriched production fuel elements failed during the period September 1, 1964 through November 30, 1964. The rupture data are presented in Table I. Seventeen of the failures were caused by corrosion, either as hot spots or as localized ledge or groove corrosion. Under the "Quality Certification Program," about fifteen to twenty ingots are canned each month, maintaining ingot identity, and obtaining pre- and post-irradiation data on one reactor charge from each ingot. This program has shown that correlations exist between the external bond count, in-reactor warp, and the frequency of hot spots and non-uniform corrosion. Some of the significant data are shown in Figures 1 through 6.

The trends on external bond count data and in-reactor warp are presented in Figures 1 and 2, respectively. It should be noted that both were at a minimum in early 1963 and at a maximum in early 1964. These trends were reflected as very good fuel performance in the second and third quarters of 1963 and as an unusually high rupture frequency during the second quarter of 1964. The similarity of the two curves suggests common causes or controlling factors for bond count and warp, and also show that the current NLO process is basically capable of producing a very satisfactory core. A series of production tests is being planned in an attempt to improve fuel performance without making any major process or equipment changes at NLO.

A relationship between external bond count and fuel performance is shown in Figure 3. In this graph, the measure of fuel performance is based on the number of lots classified as "rupture prone." These are lots which experience more than one corrosion failure, one corrosion failure at less than fifty per cent of goal exposure, or a failure induced by a manufacturing (canning) defect, and experience has shown that such lots are substandard. The data do not include dingot or test material.

Figure 4 shows the effect of warp on rupture resistance. In this graph, the intrinsic fuel quality is expressed as relative rupture resistance and the warp is expressed as approximate average warp and as observed minus expected (O - e) warp which is corrected for reactor effects on fuel stability.

Figures 5 and 6 show the effect of warp and temperature on the frequency of hot spots and non-uniform corrosion. These data show that a change in irradiation warp of 0.7 $\sqrt{\text{mils}}$ has the same effect as a 5°C change in outlet temperature on fuel hot spot frequency, and a change in warp of 0.6 $\sqrt{\text{mils}}$ has the same effect on ledge corrosion as a 10°C change in outlet temperature.

The following conclusions can be made based on current and past fuel performance and the quality certification data:

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TABLE I
PRODUCTION RUPTURES
I&E FUEL ELEMENTS

<u>Fuel Element Type</u>	<u>Reactor</u>	<u>Per Cent of Goal Exposure</u>	<u>Rupture Classification</u>	<u>Failure Date</u>
Normal Nonbumped	B	97	Side Hot Spot	9/2/64
Normal Nonbumped	DR	102	Side Hot Spot	9/10
Normal Nonbumped	B	70	Side Hot Spot	9/18
Enriched Bumped	H	92	Manufacturing Defect	9/20
Enriched Bumped	H	83	Manufacturing Defect	9/22
Enriched Bumped	DR	119	Unknown	9/26
Normal Watermix	DR	96	Side-Groove Corrosion	10/2
Enriched Bumped	DR	70	Side-Groove Corrosion	10/7
Normal Nonbumped	D	39	Side Hot Spot	10/8
Enriched Bumped	DR	88	Groove Corrosion	10/10
Normal Nonbumped	B	131	Side Hot Spot	10/11
Enriched Bumped	D	105	Side Unclassified	10/13
Enriched Bumped	H	104	Side-Groove Corrosion	10/14
Normal Nonbumped	DR	107	Unknown	10/19
Enriched Nonbumped	C	62	Side Hot Spot	10/21
Normal Nonbumped	B	70	Side Hot Spot	10/24
Enriched Nonbumped	C	55	Side Hot Spot	10/24
Normal Nonbumped	B	78	Unknown	11/4
Normal Nonbumped	H	58	Side Unclassified	11/4

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TABLE I (Cont'd)

<u>Fuel Element Type</u>	<u>Reactor</u>	<u>Per Cent of Goal Exposure</u>	<u>Rupture Classification</u>	<u>Failure Date</u>
Enriched Bumpered	F	102	Not Examined	11/6
Normal Overbore	C	96	Unknown	11/9
Normal Nonbumpered	D	79	Side Hot Spot	11/12
Normal Nonbumpered	B	100	Unknown	11/14
Normal Nonbumpered	D	78	Side Hot Spot	11/14
Normal Nonbumpered	D	84	Side Hot Spot	11/23
Normal Nonbumpered	H	89	Side Hot Spot	11/25
Normal Nonbumpered	D	88	Side Hot Spot	11/29

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CANNING DATE vs. EXTERNAL BOND
(ALL MODELS)

FIGURE 1

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CANNING DATE vs. $\sqrt{\text{WARP}}$

FIGURE 2

FIGURE 3
RELATIONSHIP BETWEEN THE
NUMBER OF RUPTURE PRONE
LOTS PER QUARTER AND AVERAGE
BOND COUNT (SEA QUARTERS 1963
THROUGH 1ST QUARTER 1964)

NUMBER RUPTURE PRONE LOTS = 1.48 LOG (B.C. + 1) - 3.09

CORRELATION

$r = 0.992$

NUMBER OF RUPTURE PRONE LOTS PER QUARTER

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Frequency of Hot Spots Versus
Temperature and Post-Irradiation Wep

FIGURE 2

Average Outlet Temperature (°C)

Model

$$y = b_0 + b_1 x + b_2 x^2$$

where:

y = proportion of fuel elements (per tube)
not spots

$$y = 2 \arcsin \sqrt{p}$$

x = Av, tube outlet temperature (°C)

x = Av, observed tube wep (mils)

$$b_0 = -1.560 \pm 1.204$$

$$b_1 = 0.0177 \pm 0.0053$$

$$b_2 = 0.128 \pm 0.222$$

03B Nonbopper Quality

Certification Charges

R. H. A. DR Reactors

(Discharged third
quarter 1962 through
first quarter 1964)

FIGURE 6
FREQUENCY OF GROOVE CORROSION ATTACK VERSUS
TEMPERATURE AND FUEL-TUBE CREATION WARP

BM-5300-074(3-60)
 AEC-GE RICHLAND, WASH.

Based on Data from
PT-310 and PT-432

D Reactor
 (Irradiated during
 the third & fourth
 quarters of CY 1962)

Model

$$Y = b_0 + b_1 x + b_2 w$$

where:

p = proportion of fuel elements (per tube)
 exhibiting groove attack

Y = $2 \arcsin(p)$

x = AV, tube outlet temperature (°C)

w = AV, observed tube warp (mils)

$$b_0 = -3.30 \pm 2.26$$

$$b_1 = 0.0268 \pm 0.0183$$

$$b_2 = 0.4839 \pm 0.310$$

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1. Fuel performance could be improved considerably.
2. The current NLO process is capable of supplying very satisfactory cores.
3. Fuel performance is influenced by external bond count and in-reactor warp.
4. There is an apparent relationship between bond count and warp, suggesting they have a common cause.

Residual Stress Measurements

In an effort to develop out-of-reactor tests to predict in-reactor fuel performance, it was decided to investigate the possibility of using the x-ray to measure residual stress present in a core after heat treatment. For initial testing, a plate was cut from a Hanford production core and annealed at 450 C for one hour and furnace cooled. This was intended to achieve a maximum of stress relief with a minimum of grain growth. An x-ray diffraction pattern was taken for the plate and compared with a pattern from a regular beta heat treated production core. The diffraction lines from the annealed plate were considerably sharper, especially in the back reflection region, than those from the production core. This suggested that there might be a measurable residual stress in the production core and an experiment was designed to investigate this.

The determination of stresses by means of X-rays is based on the measurement of changes in the lattice spacings. To determine this lattice change, an x-ray beam of wave length λ is diffracted from the sample and the diffraction pattern representing high angle interference lines (hkl) is recorded.

A quadratic form of Bragg equation for the orthorhombic case may be used for the necessary calculations:

$$\sin^2\theta = \left(\frac{\lambda}{2a_0}\right)^2 h^2 + \frac{\lambda}{2b_0} k^2 + \left(\frac{\lambda}{2c_0}\right)^2 l^2$$

Where θ is the diffraction angle, a_0 , b_0 , c_0 , are the lattice constants λ is the x-ray wave length and (h k l) are the Miller indices. Using an x-ray diffractometer, a direct measurement of the Bragg angle is made from which the lattice changes can be calculated.

The initial investigation was to determine if deflection of a uranium plate would show a line shift on the x-ray diffraction pattern. The uranium plate previously discussed was polished and prepared for x-ray analysis. The plate was x-rayed after 0, 5, 8, 16, 20, 26, and 30 mils deflection and a plot of the change in diffraction angle versus $\sin^2\psi$ was made. For this $\sin^2\psi$ procedure, it is necessary to make several different measurements of the x-ray peak position. In these measurements, the incident beam is inclined normal to the

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surface of the specimen and lies in a plane perpendicular to the specimen surface. The measurements in the plane allow the determination of the residual stress component in the direction defined by the intersection of the plane and the specimen surface. (This angle between the measurement direction and the measurement surface is designated ψ .) The lattice strain values at each deflection are plotted versus $\sin^2 \psi$ and the slope m of the line is used in the basic stress formula $\sigma_\psi = \frac{E}{\nu+1} \cdot m$ to calculate residual stress.

To date, x-ray patterns have been made from several deflections of the standard plate and at each deflection a pattern was made at five ψ angles. A plot of the changes in 2θ versus $\sin^2 \psi$ for each deflection is shown in Figure 7.

There are several problems yet to be solved. To determine accurate lattice constants requires measurements of the diffraction angle as near 90 degree θ as possible. Uranium lacks high intensity planes in this region with the (154) plane being the best. The (225) plane would be more desirable because of its higher angle location, but in an unannealed beta heat treated sample this peak is too broad to make an accurate measurement of the angle θ .

The initial data determined in Figure 1 shows an increasing slope with increasing deflection and gives reason to believe that residual stress can be determined in uranium by x-ray diffraction methods.

AlSi Process Development

The analysis of a statistically designed test using C5N cores with can-core annuli of 4, 6, and 8 mils and spire-core annuli of 4, 6.5, and 9 mils indicated the following:

1. A smaller can-core annuli will reduce the external UE-1 bond count.
2. A smaller spire-core annuli will reduce the internal UE-1 total bond count.
3. A smaller can-core annuli tends to reduce the external UT-4 total bad discs.

A modest size test (~ 14,000 pieces) is planned using K5N cores that have an OD 0.004-inch larger and an ID 0.004-inch smaller than the present K5N core. These test cores will weigh about 0.06 pound more and would therefore have a greater reactivity than present K5N cores.

Thoria Program

Feed Material

Thorium oxide is presently being supplied by Mallinckrodt Chemical Works and Nuclear Fuels Services, Inc. The compactability of the NFS oxide has been about equivalent to that received from MCW prior to October 30, 1964. Most of the thoria received from MCW since October 30 could not be

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compacted to 7.3 g/cc, but by blending MCW oxide with NFS material, densities of 7.5 g/cc are now being easily achieved. The compaction difficulties with the MCW oxide appear to be primarily caused by poor particle size distribution and, in isolated instances, low particle density. Greater than ninety per cent of the thorium has presented little or no difficulty in achieving the 7.5 g/cc target element core density.

Process

The conversion was made from a laboratory to a pilot-scale thorium element fabrication facility on November 16, 1964. The pilot-scale process is identical to that previously used with the exception that the sieving operation which was used to eliminate large agglomerates was eliminated as a result of improvements in the quality of the thorium received from off-site.

The pilot-scale facility has a design capacity of 800 elements per shift when operating at 85 per cent efficiency and is completely hooded. It consists of a mechanized drum unloading, continuous blending, automatic batch weighing, compaction, and welding.

Product and Schedules

Approximately 30,000 thorium target elements have been assembled since the beginning of production at Hanford with an over-all yield of 93 per cent. The major problem has been in welding, with an over-all welding reject rate of 4.5 per cent. The first load for the 100 Kg-U-233 order was charged into the fringe of KW Reactor on September 30, and the last load will be charged on approximately April 1 in the KW core. The charging for the 30 Kg order will begin on approximately February 15 and will be completed on May 15. It is currently planned to charge thorium in the fringe tubes of KE, KW, C, B, and D Reactors and the central zones of KE, KW, and D Reactors. All of the thorium will be discharged by July 15, 1965.

Thorium Development

A development program for increasing core density to 8.5 g/cc is in progress at Hanford. A number of particle size distributions formed from sol-gel thorium with no +6 mesh particles have been found to be easily compactable to 8.0 g/cc. The resultant change in dissolution characteristics and the over-all economics for higher densities have not yet been determined.

Self-Support Program

The out-of-reactor tests and production capabilities studies on the arch self-support rail have been completed, and were favorable. Production capabilities studies have shown an increased welding reliability with the seam weld concept used on the arch rail as opposed to the spot weld used on the bridge rail, and increased production throughput may be realized by complete automation of the rail feeding operation and reduced welding time.

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Ten tubes of arch rail fuel have been in K-West Reactor since October, and no unexpected problems have occurred. Seventeen thousand (17,000) pieces of K5 fuel with arch rails were charged in K-West Reactor for the thoria core loading in November, and an additional 39,000 pieces of arch rail fuel is being prepared for the January core loading of K-West Reactor. Arch rails are being used on the enriched driver fuel in the K Reactor core loadings of thoria to maintain sufficient cooling water flow through the reactor.

Technically, there is only one problem to be solved before a complete K-Reactor transition from bridge rails to arch rails can be effected. The reduced hydraulic resistance of the arch rail will increase the total water flow throughout the reactor, and the capability of the discharge system to handle the increased flow has not yet been determined.

N-RD PRODUCTION FUELS

N-Reactor Fuels: Production Status

During the third quarter of CY 1964, 135.5 tons of uranium were coextruded with a uranium utilization of 74.3 per cent. This utilization figure represents an increase of 4.4 per cent over the previous quarter.

The major Shop problems during the third quarter are reflected in the following major reject categories:

1. End welding rejects, which were primarily associated with the ID of the inner tube, resulted from excessive weld overhang. The reject rate increased from 3.2 per cent in the second quarter to 9.4 per cent in the third quarter. By the end of the third quarter, the reject rate had been reduced to normal levels by improved operating techniques and tightening of control limits at the welding machine.
2. Warp reject rates increased during the quarter to 14.4 per cent, an increase of 5.8 per cent over the last quarter. A significant improvement has been realized in warp before straightening, but a decrease in yield after straightening resulted in an over-all increase in warp reject rate. Intensive studies are being made to detect the cause of this shift in control.
3. Mechanically-recessed ends had a defect rate of 5.5 per cent for the third quarter. During September, a process of lathe hogging of end recesses replaced part of the conventional chemical milling process and provided a substantial cost saving in processing time and scrap recovery. During startup of the process, however, a problem was encountered in mechanical damage to the cladding by the hogging tool. The defect cause was corrected and the defect rate reduced from a startup high of 21 per cent to a month-end rate of 0.7 per cent.

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REACTOR FUEL DEVELOPMENT

N-Reactor Fuels: Development

Process Development

Alternate Uranium Compositions

The single-tube driver fuel element design will operate at higher volume mean uranium temperature and higher exposure than the current tube-in-tube design. The more severe operating conditions of the driver fuel elements require development of a uranium fuel material that is more resistant to swelling than the current composition of 150 ppm Fe + 100 Si. Fabrication development, material evaluation, and irradiation testing programs are in process for the selection of fuel materials capable of satisfactory performance under more severe conditions.

Fabrication of the driver fuel elements for the coproduct demonstration irradiations is essentially complete. Approximately 1/4 of the billets were made with British composition uranium (350 ppm Fe, 800 ppm Al). This was the first extensive production experience with coextrusion of these alloys at Hanford. No fabrication difficulties were noted; 10 per cent higher upset and running pressures were needed in coextrusion because of increased stiffness of this alloy.

Some losses in coextrusion bond strength were noted using the standard alloys. These billets were the same outside diameter as the standard N-outer billet, but because of the thicker wall in the coextruded fuel, the coextrusion reduction ratio was 15.3:1 instead of the 18.9:1 obtained with N-outer fuels. The weaker bonds were attributed to this low reduction ratio because metallurgical examination of specimens did not reveal any indications of bond separation or foreign material. After beta heat treatment of the fuels, diffusion occurred across the bond. Future material orders for coextrusion of driver tubes will be designed to utilize a new 8.1 inch coextrusion liner instead of the 7.5 inch liner. As a result, the reduction ratio will increase to 17.5:1. It is expected that greater bond strength will be attained.

A fuel element design change was made in these thick-wall driver elements. It was postulated that the thicker uranium wall and its higher operating temperature would place a greater shear stress at the cap-core junction due to the thermal expansion differences between uranium and zircaloy. Therefore, the cap design selected had a V-bottom cap that was fitted into a corresponding notch in the uranium recess. This end closure design also required that a brazed joint exist between the cap and the core as well as between the cap and the clad. Difficulty was encountered in obtaining braze metal flow and bonding between the ID face of the chevron cap and the uranium. The nature of the induction heating equipment used for end closure brazing relies on some heating by conduction from the OD of the uranium to the ID and the more massive fuel acted as a large heat sink. The problem was

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alleviated by increasing the heat input to the uranium by lowering the induction coil and by holding the assembly at temperature for 100 seconds instead of 90 seconds. However, indicated yield losses on the order of 15 per cent were obtained, and more development on optimum braze conditions will be undertaken before production is converted to this fuel design. (A parallel development requirement will be improvement of the nondestructive testing technology for detecting the bond between the cap and the uranium. Because the faces of the V-notch are not normal to the axis of the fuel, it is necessary to impinge the ultrasonic pulse through the cladding below the cap at an angle to the cladding that results in a pulse being normal to the sides of the recess. This geometry of test poses additional problems in crystal alignment.)

Plans have been made for fabricating and reactor charging up to 100 tubes in N-Reactor with simulated driver tube fuels during 1965. These fuels will be made from various alloys of uranium. The simulated driver tube fuels are fabricated from 0.947 per cent enriched uranium with an aluminum cored zircaloy-clad flow splitter.

Maximum volume mean fuel temperatures for the simulated driver tubes are about 600 C (i.e., 150 to 200 C higher than the current tube-in-tube geometry).

The following uranium alloys have been selected for the initial testing:

<u>Alloy</u>	<u>Iron, ppm</u>	<u>Silicon, ppm</u>	<u>Aluminum, ppm</u>
1	350	---	800
2	250	350	---
3	250	250	250
4	150	350	---
5	350	350	800
6 (Control)	150	100	---

Eight test coextrusions have been made from alloy billets to determine acceptable uranium fabrication routes. The billets were made from 11-inch castings that had been primary extruded to seven-inch diameter billets. Some of the material was heat treated in ingot form, some in billet form, and some was not heat treated. Coextrusions made from material that was not heat treated in either billet or ingot form had badly wrinkled claddings on both the OD and ID, apparently from the residual dendritic structure of the ingot. Heat treating of billets from the same ingots eliminated this problem. All of the test coextrusions made from the seven-inch diameter billets had lower strength bonds that are attributed to the lower reduction ratio.

The plan for the initial reactor charge of 50 columns of simulated driver tubes is to use 13-inch ingots primary extruded to a 7-1/2-inch diameter billet, and coextrude from an eight-inch liner. By using these geometries, primary extrusion and coextrusion reduction ratios will be maintained at values known to be successful for N-outer

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fuel coextrusion. In addition, all uranium will be heat treated in either ingot or billet form.

Experimental work is underway on test quantities of uranium to determine potentially more economical and/or higher quality primary uranium fabrication routes. These tests include evaluation of (1) billets cast directly to size and beta heat treated and (2) 11-inch diameter solid ingots that will be upset to 13-inch OD and pierced.

Fuel compositions amenable to current N-fuel fabrication processes are being prepared for fuel element irradiation tests. Materials containing 250 ppm Fe + 250 ppm Si + 250 ppm Al, 250 ppm Fe + 350 ppm Si, and 350 ppm Fe + 800 ppm Al were coextruded to the driver-tube size and sections are being processed in ex-reactor material evaluation tests in both the beta and gamma heat treated conditions.

Material containing 179 ppm Fe + 250 ppm Si (Billet No. 2036385-extrusion X-93) was coextruded to N-outer fuel stock for study of the structural effects of heat treatment and determination of recrystallization behavior. Sections of the material in the as-extruded, beta heat treated, and gamma heat treated conditions were cold worked to 21.4 per cent area reduction and heated at temperatures from 400 to 625 C for 20 hours. Hardness measurements were made and the microstructure was examined for degree of recrystallization and distribution of intermetallic compounds. The temperatures at which 50 per cent recrystallization was observed and 50 per cent decrease in hardness (i.e., half-hard) was measured are shown in Table II.

It is seen that increases in recovery and recrystallization temperature are produced by increasing silicon content and by heat treatment.

TABLE II

Recrystallization and Half-Hard Temperatures*
of Uranium Containing Iron and Silicon Additions

<u>Composition</u>	<u>Prior Treatment</u>	<u>Recrystallization Temperature</u> (°C)	<u>Temperature for 50 Per Cent Decrease In Hardness (Half-Hard)</u> (°C)
179 ppm Fe + 250 ppm Si	Coextruded	475	430
" " "	Beta Heat Treated	490	465
" " "	Gamma Heat Treated	500	455
159 ppm Fe + 418 ppm Si	Coextruded	475	435
" " "	Beta Heat Treated	510	455
" " "	Gamma Heat Treated	535	500

*21.4% cold work, 20 hour anneal, 50% recrystallization estimated from microstructure.

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Metallographic examination of samples of both materials after the annealing treatments showed (1) the gamma heat treatment was more effective in obtaining complete solution of Fe and Si compounds than the beta heat treatment, and (2) for both heat treatments and materials, progressive precipitation and growth of intermetallic compounds occurred at annealing temperatures above 425 C.

Tensile tests and tensile stress cycling tests were made at 400 C on samples of the uranium alloys that had been irradiated in the K5E-5 test fuel design. The alloys were 150 ppm Fe + 100 ppm Si and 400 ppm Fe + 800 ppm Al and were tested in the beta heat treated condition. The purpose of the tests is to study the structural effects of cyclic deformation and to determine differences in mechanical behavior of the two materials. Standard tensile properties of the materials at 400 C are shown in Table III.

TABLE III

Tensile Properties of U + 150 ppm Fe + 100 ppm Si
and U + 400 ppm Fe + 800 ppm Al at 400 C

<u>Uranium Alloy (ppm of Additive)</u>	<u>0.2% Yield Strength (1000 psi)</u>	<u>Ultimate Strength (1000 psi)</u>	<u>Elongation (% in One-Inch)</u>
150 Fe + 100 Si	33.3	40.2	19.8
400 Fe + 800 Al	42.3	51.0	14.2

Cyclic tests were run at 25,000 ± 5000 psi, 27,500 ± 5000 psi, and 30,000 ± 5000 psi at 400 C at a cycling rate of six to seven cpm to a total of 300 cycles. The 150 ppm Fe + 100 ppm Si material at 30,000 ± 5000 psi was strained high enough into the plastic range that after 14 cycles it was continuing to yield just below 35,000 psi and the test was terminated. Total deformation as a result of cycling is indicated below.

<u>Uranium Alloy (ppm of Additive)</u>	<u>Number of Cycles</u>	<u>Nominal Stress (1000 psi)</u>	<u>Total Strain (% in One-Inch)</u>
150 Fe + 100 Si	300	25.0	0.5
" " "	300	27.5	0.8
" " "	14*	30.0	6.7
400 Fe + 800 Al	300	25.0	0.2
" " "	300	27.5	0.2
" " "	300	30.0	0.2

*Test Terminated

The density and structure of the material in the gage section is being studied for evidence of void formation. Longitudinal sections of the

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fracture region of the samples tested at 400 C have been examined metallographically. Voids formed in both materials do not appear related to grain intersections. A noticeable reduction in void size was apparent in the 400 ppm Fe + 800 ppm Al material.

The fuel element tests are being supplemented by a program of capsule irradiations to distinguish the effect of alloying additions upon fabrication, corrosion resistance, and swelling performance. Material evaluation and heat treating effects are included in these studies. These evaluations are in progress and irradiation assemblies are being prepared. Irradiations will begin in March 1965.

End Closure Development

Development work continued on fabrication of an unbonded TIG braze 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 recast 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 the closure required simultaneous development of a source of beryllium-zircaloy wire. The wire had been made by hot swaging cast beryllium-zircaloy clad in stainless steel at about 750 C. Welding development has been essentially completed on the heavy wall driver tube geometries and N-outer tubes. Fuels are being prepared for reactor testing.

Ellipticity of cladding, and particularly lack of concentricity of the two claddings, are problems hindering precise fitting of end caps in the chemically-milled cavities in the development of the TIG braze closure. Tooling has been designed, fabricated, and demonstrated for sizing and truing the free standing cladding (after chemical milling) in the end closure region. The outer cladding is sized by spinning, and the inside cladding is sized and trued

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(with respect to the outer cladding) by a device which approximates tube rolling. Initial experiments have produced excellent cap fit-up resulting in welded closures that have consistent welding conditions.

Support Fabrication Development

Attempts to cold form the "Tee" cross-section N-fuel supports in a 100-ton crank press were unsuccessful, and resulted in cracked parts and broken die inserts. The dies were modified for hot forming using a 2.5 KVA resistance welder for a heat source. The welder is marginal in capacity and does not heat the blank above 640 C for forming. The 2.5 KVA unit has been replaced by a 50 KVA unit which has ample heating capacity. No accurate determination of forming temperature has been attempted but the parts are formed in the beta phase region. Die quenching appears to be an important part of the forming cycle for mechanical stripping purposes, and the crank press control has been modified to provide this step in the forming cycle.

Forming the supports in the crankpress provides better dimensional stability than was observed for those parts formed in the 600 KVA welder.

Irradiation Experience

KER Loop Irradiation Results

"N" Coproducer Test

Fuel swelling of thirteen 1.25 per cent enriched driver elements of KER irradiated coproducer assemblies was determined by pre- and post-irradiation bulk density measurements. The drivers are coextruded, zircaloy-2 clad, tubular elements of 2.29 inch OD and 1.44 inch ID with fully brazed end closures using V-bottomed caps (i.e., tapered or Chevron design). Fuel was of the standard composition, i.e., nominal 150 ppm Fe and 100 ppm Si. Table IV summarizes the fuel exposure and fuel swelling of the 13 drivers in the test charge.

TABLE IV

Summary - Exposure and Swelling of KER Tested
Coproducer Driver Elements

<u>Position</u> Upstream	<u>Exposure</u>		<u>Fuel Swelling</u> %ΔV
	<u>MWD/T</u>	<u>a/o BU</u>	
1	500	0.006	0.25
2	800	0.09	0.24
3	1000	0.12	0.55
4	1200	0.14	0.58
5	1550	0.18	0.92
6	1850	0.21	2.31

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Table IV (Cont'd)

<u>Position</u> Upstream	<u>Exposure</u>		<u>Fuel Swelling</u> % ΔV
	<u>MWD/T</u>	<u>a/o BU</u>	
7	2000	0.23	3.27
8	1900	0.22	3.32
9	1800	0.21	2.74
10	1500	0.17	1.18
11	1200	0.14	0.65
12	1000	0.12	0.55
13	650	0.08	0.37

Radiometallurgical examination has been completed upon the maximum exposure element that was in position 7. There was a considerable amount of small, grain-boundary-torn voids in the cooler fuel (~400 C) regions adjacent to the clad surfaces. The fuel in these regions displays a "hashed" or swirled microstructure that is characteristic of uranium irradiated in the 350 to 450 C temperature range. There is more tearing on the OD side of the fuel wall than on the ID side (where it seems that the fuel would be under a tensile stress during operation). The hotter central zone of the fuel (~530 C) displays a defined metallographic structure that is somewhat more heavily twinned than the unirradiated fuel. There is considerable spherical porosity (emissary) resolvable at 1000X that is clustered and oriented to some degree with the deformation bands or twins of the microstructure.

Both end closures have been metallographically examined. The tapered or Chevron design (90° male taper on cap with matching 90° female groove in fuel) appears to have performed well with no indications of instability at the various metal junctures of the closure. A crack was observed in the fuel between the clad wall and the tapered side of the cap but there was no accompanying deformation in the clad. It is possible that the crack origin was in post-irradiation handling and not an irradiation effect.

Dimensional Changes of Prototype N-Reactor Fuel Elements

Dimensional measurements of N-Reactor fuel element components and their prototypes have been made as part of the program to understand and evaluate the effect of the irradiation environment upon performance. Pre- and post-irradiation measurements have been made using machines which determine deviations in the fuel element components from standard right cylindrical tubes. Two series of measurements used to characterize the components are the wall thickness and outer surface deviations in a plane normal to the tube axis. The measurement data are processed by a computer program to yield useful information on performance.

Most of the components of two KER loop charges of prototype N-fuels have been pre- and post-measured and evaluated. The percentage change in the cross-section of the wall of the outer component at

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ten stations of measurement have been calculated from the pre- and post-irradiation measurement data. Fuel swelling in the outer components, determined from pre- and post-irradiation bulk density measurements, are in very good agreement with those calculated from the wall thickness changes. The measurement data shows a significant swelling gradient along the length of the component in many cases. A similar comparison has been made of swelling behavior of the inner component. The agreement is not as consistent in the case of the inners, however, because of a relative lesser precision of the wall thickness measurements (± 0.002 inch).

The changes in the outside and inside radii of the outer component have also been calculated from the data. It was shown that the inner and outer radii changes follow the fuel swelling behavior with the inner decreasing and the outer increasing along the length of a test column to the point of maximum fuel swelling. The maximum increase in the outer radius of an outer component was 0.007 inch while the maximum decrease in the inside radius was 0.004 inch. The results further indicate a stable behavior of the inside bore of the outer component (i.e., no buckling up to 3.6 vol.% fuel swelling). Changes in the outside radii of the inner component were also measured and a maximum increase of 0.006 inch was determined.

Maximum "single-throw" warp calculated from the measurements for an 18-inch span were 0.016 and 0.012 inches for the outers and the inners, respectively. For the outer component, the magnitude of the change in the warp vector appeared to be a function of the position in the test charge. No such relationship was indicated for warp in the inner components, however.

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 and Al additions (400 ppm and 800 ppm, respectively) have been irradiated in high-temperature, high-pressure (KER) loops. A test charge of eight elements irradiated to 1530 MWD/T average was described in the previous report. A second test charge of eight elements has been irradiated to 2700 MWD/T average and evaluated.

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 (i.e., identical irradiation histories). Fuel swelling data, exposures, and operating temperatures of the second test charge are summarized in Table V.

TABLE V

Summarized Operating Temperatures, Exposures,
and Fuel Swelling in KSE-5 Uranium Alloy Fuel Elements

Position	Fuel Alloy	Vol. Mean T C	Fuel Exposure		Fuel ΔV%	Calc. Theor.	
			MWD/T	a/o Bu		Min.	ΔV
Upstream							
1	Fe + Al	410	2200	0.25	0.58	0.76	
2	Fe + Si	415	2200	0.25	0.81	0.76	

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Table V (Cont'd)

<u>Position</u>	<u>Fuel Alloy</u>	<u>Vol. Mean T C</u>	<u>Fuel Exposure</u>		<u>Fuel ΔV%</u>	<u>Calc. Theor.</u>	
			<u>MWD/T</u>	<u>a/o BU</u>		<u>Min.</u>	<u>ΔV</u>
<u>Upstream</u>							
3	Fe + Al	445	2375	0.28	0.75	0.83	
4	Fe + Si	475	2525	0.29	1.59	0.87	
5	Fe + Al	495	2650	0.31	1.13	0.92	
6	Fe + Si	510	2675	0.31	2.22	0.93	
7	Fe + Al	520	2750	0.32	1.17	0.96	
8	Fe + Si	518	2650	0.31	2.02	0.92	

Downstream

The uranium fuel containing Fe and Al additions swelled about as much as might be anticipated to accommodate the solid fission products, 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 (emissary) whereas the Fe and Al-bearing fuel is relatively free of such porosity. Post-irradiation fuel density determination made in the Radiometallurgy facility 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.

Post-irradiation annealing studies have been started on samples of each of the irradiated fuels to further characterize their relative behaviors. Swelling resulting from annealing at 625 C for 100 hours are tabulated below. This swelling is apparently the result of fission gas precipitation in the fuel material, but electron microscopy of the annealed samples is yet to be done.

<u>Sample</u>	<u>Irradiation</u>	<u>Annealing ΔV</u>	<u>Total ΔV</u>
	<u>ΔV at ~0.31 a/o BU</u>	<u>100 hrs. @ ~ 625 C</u>	
U with Fe & Si	2.2%	1.1%	3.3%
U with Fe & Al	1.2%	0.7%	1.9%

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Current Reactor Fuel Development

Alternate Process Development

The hot die sizing (HDS) process is being developed as a replacement for existing AlSi dip brazing fuel element fabrication process for "old" Hanford reactors. Aluminum cladding is diffusion bonded to electronickel-plated uranium cores by the application of heat and pressure. Preheated fuel assemblies are passed through a ring die to bond the external surface. Concurrently, the piece is passed over a die plug to bond the internal surface. End bonding is accomplished by applying heat and pressure to the fuel element ends.

Development of the hot die sizing process has proceeded to the point of increasing quantities to augment production. The HDS Pilot Plant is scheduled to be ready for operation January 1, 1965 for the purpose of producing fuel at a rate of 5000 pieces per month. The first sustained operation of the automatic nickel plating machine was a one-week run late in October 1964. During the next run, design plating rate was reached. Installation of a high capacity preheat furnace and a more versatile sizing press is nearing completion. Original development preheating and sizing equipment was used for producing initial production test material and sizing approximately 2200 "O"-size and "K"-size HDS fuel elements.

Post-irradiation measurement of initial hot die size production tests revealed an unforeseen growth behavior. Hot die sized fuel increased 0.008 inch more than corresponding AlSi brazed fuel. The next two production tests behaved similarly. For this reason, emphasis of process development work was shifted to determine the cause of increased growth during irradiation. Metallurgical examination and dimensional testing indicated that both core heat treating and end bonding could be contributing to the problem.

A fourth production test showed a slight improvement in the growth problem. Hot die size fuel elements heated by induction during end bonding had 0.002 inch less growth than pieces end bonded in the resistance heated end bond press. Average changes in the test and control pieces are shown below:

	AlSi Brazed Controls	Induction Heat End Bonding	Resistance Heat End Bonding
OD-1 3/4 inch from base	3.0 mils	6.2 mils	7.3 mils
OD-2 Center	5.8 mils	12.5 mils	14.5 mils
OD-3 3/4 inch from cap	6.5 mils	11.6 mils	12.9 mils
ID-1 3/4 inch from base	3.0 mils	13.3 mils	13.9 mils
ID-2 Center	3.0 mils	12.3 mils	12.3 mils
ID-3 3/4 inch from cap	2.8 mils	11.7 mils	12.5 mils
Warp	2.6 mils	2.5 mils	2.4 mils
Weight Loss	1.2 gm	1.2 gm	1.4 gm

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As was experienced in previous tests, a large percentage of all fuel in this test (~ 52 per cent) had a double-bend profile. This characteristic as well as above-normal warp was common to all test fuel and is believed to be a function of the striped loading used for this test.

The latest hot die size production test discharged was designed to determine the effect of end bond pressure and core diameter upon fuel dimensional behavior during irradiation. End bond pressure was varied between 3-3/4 and 4-1/4 T/in.². Core diameter was reduced by 0.026 inch to remove a layer of high growth index material produced by thermal gradients during heat treating. The fuel elements have been measured and the data are being evaluated. External diameters of AlSi brazed and HDS fuel differ with some indications that the HDS growth was not as great as in previous tests. Statistical significance of minor differences between large and small cores and the end bond pressure variation have not been established.

The production test designed to test the effect of core heat treatment upon irradiation dimensional stability is scheduled to be discharged December 21, 1964. Cores from alternate rod positions were oil (hollow) and water (solid) quenched. Portions of each type were stress relieved by alpha annealing and AlSi brazing pretreatment.

Twelve charges containing hot die size fuel elements clad with tubing components are scheduled to be charged in C Reactor during the December outage. This test consists of alternating AlSi dip-brazed and hot die sized fuel elements.

Procedures for producing overbore fuel have been developed. Reactor grade overbore fuel clad by the hot die sizing process is scheduled to be produced late in December 1964.

A flow charge of integral ribbed fuel produced by the hot die sizing process is being fabricated. Integral ribs were produced by sizing fuel assembled with smooth wall tubing through a slotted die. The ribbed fuel elements will soon be ready for flow testing.

High Silicon-Aluminum Alloy Cladding

Corrosion testing of high silicon-aluminum alloy (six to ten per cent Si) for hot die size diffusion bonded fuel elements is continuing. Complete evaluation of corrosion characteristics will not be available for several months due to the time required to test representative samples of the alloys in reactor process water. A tentative process for fabricating hot die size fuel clad with high silicon aluminum alloy has been established.

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