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HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

April 8, 1963

TITLE

REPORT TO THE WORKING COMMITTEE FROM
THE GENERAL ELECTRIC COMPANY, HAPO

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REPORT TO THE WORKING COMMITTEE FROM
THE GENERAL ELECTRIC COMPANY, HAPO

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By Authority of PR 24 (CG-FR-2), April 8, 1963
D. S. Lewis, 3-18-64
By Jerre Moley, 3-31-64
Verified By L. K. Hanson, 4-4-64

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1.1

PRESENT REACTOR FUEL PRODUCTION

Uranium Core Production Data

Quality of Uranium Cores

In order to assure uniformity of incoming cores, a control chart system is under development for monitoring:

1. Chemical analysis, dimensions, isotopic measurements, grain size, and other data presently supplied by the feed site.
2. Equivalent data available from receiving inspection at HAPO.
3. X-ray diffraction information from HAPO inspections.
4. HAPO metal quality and UT-2 grain size rejects.

It is believed that this "control chart" system of auditing large amounts of data will be of value in determining changes in the product which are due to both intentional and random process variations. In order to maintain this system, extensive use is being made of electronic data processing equipment. An extensive effort is being made to assure that both HAPO and NLO supply input data promptly and with as few errors as possible.

Although the data processing systems needed for this information will not be available until the 2nd Quarter CY 63, some of the data generated over the last three months have been subjected to this type of analysis, and no significant trends were detected. However, in the interest of continuity with previous reports, the following specific areas were analyzed:

1. Visual Inspection of Virgin Cores - Visual reject rates averaged less than .3% with individual lots varying from as high as 2% to no rejects.
2. UT-2 Grain Size Rejects - The reject rate for this period was approximately 1.2%, which is also consistent with past performance.
3. Out-of-Specification Material - One ingot was discovered by the HAPO receiving audit to be significantly out of specification. This enriched ingot--number 41621--contained over 400 ppm iron and was shipped due to a mix-up in analytical samples at NLO. Unfortunately, the cores from this ingot had been canned before the error was detected, which resulted in a costly recovery program.
4. Uranium Grain Size - The average grain size for normal cores during the period 12/1/62 to 3/1/63 was a 4.7; while that for enriched was a 5.7. This represents no change from last

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quarter's figures for the enriched, and a tendency toward finer grains in the normal material.

5. Hydrogen Analysis - The average hydrogen analysis on cores received during the period 12/1/62 to 3/1/63 showed 1.3 ppm for non-outgassed material and 0.5 ppm for outgassed. No differences were detected between enriched and normal cores or between models, with the possible exception of overbore cores which appear to have a lower hydrogen concentration.

High Phosphorous in Enriched Cores

It was discovered during this quarter that the Winlo process at NLO was introducing abnormally large amounts of phosphorous in a large number of enriched ingots. Since the normal phosphorous level in this material is less than 10 ppm, and there was previously no way in which phosphorous could be introduced into the process, no specification for phosphorous was set. With the possibility existing that material with up to 300 ppm might be received, a temporary limit of 75 ppm was imposed, and later, when it was observed that it would not be possible due to short inventories to meet production commitments with this specification, relaxed this to 100 ppm. It is expected that the 100 ppm specification will remain in effect until the approximately 100 tons of contaminated material presently at NLO is exhausted.

Vacuum Outgassing of Uranium Cores

Vacuum outgassing, which had been initiated for the purpose of decreasing the hydrogen level in enriched cores, was discontinued on bulk production quantities pending the results of a specially designed production test. This action was taken when it was observed that some outgassed cores had an extensive build-up of impurity on the internal diameter. This impurity was identified by X-ray diffraction as UO_2 . Figures 1 and 2 are photographs of etched sections of one of these cores, which clearly shows the I.D. oxide build-up. Figure 3 is a polished section which shows the thickness of the layer. Figure 4 is from a core after treatment in the production HNO_3 bath. It can be seen that although most of the oxide has been removed, an appreciable amount still remains.

Oil Quenched Cores

Although delivery of oil quenched cores has been very slow during the first quarter of CY 63, this material continues to be reactor charged under a process improvement transition authorization which authorizes the loading of up to 50 tons per month in B, D, DR, or F reactors.

The results of the initial oil quenched reactor test have been published, and the indications are that oil quenched cores, which

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Figure 1. TRANSVERSE SECTION FROM VACUUM OUTGASSED
CORE, INGOT NUMBER 39707. 2.5X

Note UO_2 film in the bore which, except
for local discontinuities in the oxide,
prevented the deep HCl etching of the
uranium.

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Figure 2. LONGITUDINAL SECTION FROM VACUUM OUTGASSED
CORE, INGOT NUMBER 39707. 2.5X

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U UO₂

Figure 3. LONGITUDINAL SECTION FROM CORE, INGOT
NUMBER 39707. THIS IS IN THE AS-
POLISHED CONDITION. 250X

The UO₂ is about .0016 inch thick,
maximum.

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Figure 4. LONGITUDINAL SECTION FROM CORE, INGOT
NUMBER 39707. THIS IS A SECTION THROUGH
THE CORE AFTER THE PRE-CANNING HNO_3 ETCH. 250X

Note small remnants of UO_2 film remaining
in localized areas.

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were irradiated to approximately 800 MWD/T ton, are substantially more stable during irradiation than the water quenched material. Table 1.1 is a summary of the results of this test.

Table 1.1

<u>Characteristic</u>	<u>Number of Pairs</u>	<u>Average Difference (W-O)</u>	<u>95% Confidence Limits on (W-O)</u>
Δ OD-1	159	-1.24 mils	$\pm .57$
Δ OD-2	159	2.28 mils	$\pm .79$
Δ OD-3	159	-0.36 mils	$\pm .54$ (not significant)
Δ ID-1	159	3.81 mils	$\pm .52$
Δ ID-2	100	3.34 mils	$\pm .43$
Δ ID-3	159	3.47 mils	$\pm .41$
Δ Length	159	6.25 mils	± 5.28
Δ Volume	100	.0144 in. ³	$\pm .0193$ (not significant)
Warp	159	0.38 mils	$\pm .20$
TFC	159	4.08 mils	± 1.11

The warp averages were 4.71 mils for oil and 6.11 for water.

No difference was detected in the degree to which the two groups showed surface roughening or "bumping." Bumping was visually detected on 21 of 197 oil quenched and 20 of 184 water quenched elements.

In view of these promising results, advanced testing and studies aimed at thoroughly evaluating the oil quench treatment are being given high priority. This consists of:

1. A study to determine if the change to oil quenched cores is economically sound.
2. A repeat of the above test utilizing enriched cores.
3. An extensive heat treat test which will evaluate the effects of feed site variables on both oil and water quenched cores.
4. A production test to evaluate induction versus salt bath heating of cores during heat treatment. The National Lead Company believes that induction heating has certain advantages over the present salt bath method, and the optimum heating method should be decided upon at the same time the quench medium is finalized.

It is presently anticipated that these studies can be completed before February, 1964, and a proposal regarding both oil quenching and induction heating will be made at that time.

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Fuel Performance

Rupture Summary

Fourteen (14) natural and eleven (11) enriched standard production fuel elements failed in the Hanford reactors during the period from January 1, 1963 through March 31, 1963. Irradiation and rupture classification data for these failures is summarized in Table 1.2. In addition, one rupture was incurred at C reactor in natural self-supported fuel; a compound-split failure at 738 MWD/T.

Rupture Performance Trends

In bumper fuel the major cause of failures continues to be non-uniform corrosion attack and, as previously reported, the failures are largely confined to the high exposure ranges beyond which little or no non-bumper fuel is irradiated. To date, no side hot-spot type failures have been incurred in bumper fuel.

In non-bumper fuel the majority of failures are of the side hot-spot type, although during the past six months, several side-other type failures were attributed to non-uniform corrosion. Rupture resistance (q) of natural non-bumper fuel during the first quarter of 1963 was comparable with that experienced during previous six months to nine months; q for the first quarter was estimated as .07 (+.10, -.05). Rupture resistance (q) of enriched non-bumper fuel declined during the first quarter, q being estimated at .15 (+.17, -.09).^{*} This difference in rupture resistance between enriched and natural may be attributable to differences in bond quality based on past experience.

Post-Irradiation Warp and Dimensional Changes

Dimensional performance data for ingot natural fuel have been extended about three months beyond those presented in the last Working Committee Report; the data now includes canning through about May 15, 1962. Post-irradiation warp remained stable at $2.4/\text{mils}$; ΔOD_2 and TFC dropped markedly; average ΔOD_2 for fuel canned during April and May was -0.5 mils; average TFC was $3.0/\text{mils}$. There were no significant differences between fuel models.

* See DPST-63-123, "Report of the Working Committee of the Fuel Element Development Committee, January 21-24, 1963," for historical trends in (q).

TABLE 1.2

PRODUCTION RUPTURES
I&E FUEL ELEMENTS

Date	Reactor	Type of Fuel Element	Exposure		Tube Power at Failure (kw)	Rupture Classification
			(MWD/T)	(% Goal)		
1/2/63	D	Natural Bumper	672	(82)	1177	Side Other*
1/6/63	H	Enriched Bumper	837	(122)	1138	Side Other*
1/9/63	H	Natural Bumper	605	(97)	1101	Side Other*
1/14/63	DR	Enriched Bumper	836	(95)	1058	Hole*
1/17/63	D	Natural Bumper	794	(95)	932	Hole*
1/17/63	D	Natural Bumper	840	(102)	1329	Hole*
1/28/63	DR	Natural Bumper	724	(128)	1102	Hole*
1/29/63	F	Natural Non-Bumper	569	(102)	1263	Side Hot Spot
1/30/63	DR	Natural - Water Mix Piece	743	(130)	1013	Split
2/2/63	DR	Enriched Bumper	963	(104)	1005	Hole*
2/2/63	DR	Natural Bumper	756	(126)	969	Hole*
2/3/63	D	Natural Bumper	927	(102)	1244	Hole*
2/6/63	KE	Enriched Non-Bumper	589	(88)	1883	Side Hot Spot
2/9/63	H	Enriched Bumper (E-N)	84	(10)	1107	Mechanical Damage
2/14/63	DR	Natural Bumper	999	(164)	1226	Hole*
2/19/63	D	Natural Bumper	878	(92)	1189	Side Other*
2/24/63	C	Enriched Non-Bumper	625	(79)	1370	Side Hot Spot
2/26/63	D	Natural Bumper	1030	(119)	1263	Hole*
3/4/63	F	Natural Non-Bumper	411	(72)	1229	Side Other
3/7/63	KW	Enriched Non-Bumper	562	(77)	1733	Side Hot Spot
3/11/63	B	Enriched Non-Bumper	288	(44)	1231	Side Hot Spot
3/11/63	H	Enriched Non-Bumper	869	(102)	1017	Side Other*
3/23/63	B	Enriched Non-Bumper	453	(58)	1243	Unclassified
3/26/63	DR	Natural Non-Bumper	380	(73)	1182	Side Hot Spot
3/31/63	H	Enriched Bumper (E-N)	584	(73)	1083	Split

*Probably caused by non-uniform corrosion attack.

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NRD PRODUCTION FUELS

Fuel-Production Status

Completed fabrication of the first reactor loading of N-fuel elements, approximately 450 tons, is scheduled for February, 1964. To date, approximately 56.6 per cent of the load has been coextruded. Thirteen per cent (50.4 tons) has been autoclaved but these inner and outer fuel elements have not been assembled and locked together. Most of the in-process inventory is backlogged at support welding (160 tons). Full-scale operation of both the support attachment and assembly steps will be accomplished during April, 1963.

Over-all yield cannot be determined at this time. It appears that material now in process will be completed with a total yield of 70 to 80 per cent. Station reject rates at the end of the first quarter of calendar year 1963 are shown below:

TABLE 2.1

N-Reactor Fuel Reject Rates- First Quarter, CY-1963

<u>Station</u>	<u>Reject Rate, Per Cent</u>	
	<u>Outer Fuel</u>	<u>Inner Fuel</u>
Bonding (Peel Test)	2.0	5.9
As-Coextruded Eccentricity	0.6	0.4
Clad Thickness Variation	7.3	0.7
Heat Treated Warp	0.8	4.6
End Closure	14.6	4.3
Support Attachment	Insufficient data	
White Oxide after Autoclaving	Insufficient data	

Chemical Composition: Phosphorus Contamination In Ingot Metal

During February 1963, Fernald reported phosphorus contamination in the 0.947 per cent enriched metal supply for N reactor. Immediate concern for the irradiation performance of N fuel elements was expressed by NRD. To date, the Fe, Si additive uranium has shown good swelling performance. It remains to be seen if the phosphorus contamination will change this aspect of in-reactor performance.

Two hundred and eleven ingots in the casting series 36,400 through 36,610 were reported to contain up to 360 ppm P. To meet production schedules, Fernald suggested a specification of 100 ppm. Based on both the unknown effect of P on swelling performance and the high cost of manufacturing Zircaloy-2 clad fuel

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elements, NRD has imposed a tentative specification of 50 ppm P. At this level of P contamination, continuity of production schedules can be maintained with minimum risk of both the amount of fuel elements with contamination and the degree of contamination.

Development work on the mechanical and physical metallurgy of the system uranium-phosphorus is described in the Research and Development section of this report. Preliminary manufacturing experience in the ingot heat treating, primary extrusion, coextrusion, and fuel element manufacturing steps was obtained by chance with ingot number 36,101 which contained approximately 170 ppm P. (The P analysis of this ingot indicates that more than 211 ingots are contaminated). No adverse effect at this level of P was immediately apparent. However, in-reactor performance and the problems that could arise from fuel element breakup during the free-fall after discharge will ultimately determine the P specification for N-reactor metal.

Uranium Billets

N billets received at HAPO have been within the specifications for metal quality and dimensions except for a few minor excesses. To improve the economics of the fabricating process at NLO, work is underway to improve both the metallurgical and dimensional symmetry of finished billets to permit use of ingots with large, as-cast, wall thickness variation.

Billets are currently fabricated from 11-inch hollow castings by alpha extruding to rough size and machining to final dimensions. Primary alpha extrusion tests have shown that chamfering of the lead end of the ingot will improve dimensional concentricity by causing the lead end of the ingot to center in the entry cone of the die.

Some ingots have been machined to improve both O.D. surface finish and concentricity. Ingots that do not meet the wall variation specification of 0.060 inch maximum are being O.D. turned to less than 0.080 inch wall variation and chamfered on the lead end. If favorable extruded dimensions are obtained, O.D. machining and lead-end chamfering of all ingots will be specified as soon as process facilities can be put in place.

Alternate plans are to cast solid ingots and drill the hole. Improved ingot metal soundness and concentricity are expected. Test quantities of this process change were coextruded in late March, 1963; initial evaluation showed favorable results.

Coextrusion Process: Inner Clad Bonding of the Inner Tube

In December 1962, unbonding of the inner clad of the inner tube was encountered. The conditions that caused this problem are not entirely understood but adequate bonding has been regained by use of 1/2-inch thick end plates at the front end of the billet assembly. The thicker copper plates have increased upset pressures about 300 tons over the 1700 tons encountered with 1/4-inch thick end plates.

The general characteristics of this bonding problem were different from those

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experienced in the past in that (a) the clad-core interface was free of contamination in the form of uranium oxides, nitrides, etc., (b) there was no apparent loss of billet vacuum during the billet preheat, and (c) approximately 40 to 50 per cent of "unbonded" coextrusions had 6 to 8-inch of good bond at both ends.

In several test coextrusions, it was shown that some improvement in bonding was obtained by machining the I.D. of the billet 0.040-inch oversize. However, this gain was not as effective as the use of the thicker copper end plate which is now routinely used in production.

Autoclave Testing

The locking clip, which fastens the fuel elements in an assembly, is attached to the inner clad of the outer tube. The clip is attached by a variation of standard projection resistance welding techniques known as spike welding. Critical control in spike welding is required to assure reproducible penetration and freedom from cracks.

Two autoclave failures, that were apparently related to defective, locking-clip welds, have occurred in separate autoclave loads of outer elements. Both failures occurred in 400°C steam at 1500 psi pressure after 9 and 19 hours residence.

The defective welds have been characterized as having excessive penetration of the inner clad with a resulting defect that allowed steam contact with the uranium. The defects were not visible; they were covered by the locking clip.

The cause of the excess penetration has been related to equipment malfunction. Correction of the equipment now permits the control of the penetration zone within 2 mils at 10 mils average depth.

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3.1

FUEL RESEARCH & DEVELOPMENT

N-FUEL DEVELOPMENT

Process Development

Materials

Uranium - Phosphorus Impurity

Phosphorus, in quantities to 400 ppm at the ingot stage, was introduced into the metal for N-reactor and the enriched I&E fuel due to a change in scrap recovery processes at NLO. The effect of this additional contaminant on the behavior of uranium as a fuel material is of immediate concern. Metal containing three levels of phosphorus, 430, 280, and 135 ppm, have been examined. The structure has been interpreted as containing a eutectic, composed of uranium and the phosphorus compound UP. The composition of the eutectic is estimated to be 1000-1500 ppm phosphorus. A melting point depression of about 10 C was observed. There appears to be little, if any, solid solubility of phosphorus even in the gamma phase. In ingot metal containing carbon, the carbide enters the eutectic reaction, possibly as a ternary eutectic (U-UP-UC). Within the eutectic network the phosphide appears to have nucleated the carbide formation similar to the nucleation of the primary carbide by uranium nitride. Photomicrographs illustrating these features are shown in Figures 1 through 5. Initial examination of NIE sections containing 170 ppm phosphorus in the as-coextruded, beta heat treated, and brazed and beta heat treated conditions has shown that the eutectic network is dispersed during the primary extrusion and coextrusion steps. In the braze heat affected region and in beta heat treated sections there is no evidence of solubility of the phosphide in either the beta or gamma phases. The resulting structure is one of additional inclusions, oriented similar to the carbides, but generally rod or plate-like in shape and smaller than the carbide. The effect upon mechanical properties, etching characteristics, bond strengths, and irradiation behavior remains to be determined.

Uranium - Alternate Compositions

Two primary extrusions and five coextrusions of normal uranium alloyed with 640 ppm of aluminum, 400 ppm of iron, and 110 ppm of silicon have been made in the N-outer tube geometry. No significant fabrication problems were

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100X
As Polished

500X
As Polished

100X P.L.
Electrolytically Etched

FIGURE 1

Derby Metal-430 ppm Phosphorus - Metal as Reduced

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FIGURE 2

Ingot 36421 (NOE) 280 ppm Phosphorus Average. All 100X As Polished.
Note Change in Eutectic Dispersion and UC Size with Ingot Location
(Solidification Rate).

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Ingot 36421 (NOE) 280 ppm Phosphorus Average. All 500X As Polished.
Note Small UC Inclusions within Eutectic.

FIGURE 3

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FIGURE 4

Ingot 36407 (NIE) 135 ppm Phosphorus Average. All 100X As Polished.
Note Decrease in Amount of Eutectic at this Lower Phosphorus Content.

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FIGURE 5

Ingot 36407 (NIE) 135 ppm Phosphorus Average. All 500X As Polished.
Note Small UC Inclusions in Eutectic.

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encountered. The extrusion constant during coextrusion was increased compared to normal iron-silicon material ("K" running 16.6 to 19.1 TSI). The extrusion constant, K, was decreased slightly with increase in extrusion preheat time. Evaluation of one coextrusion indicated good bonding and wall thickness uniformity with excellent clad thickness control (± 0.003 inches maximum).

Sections of the coextruded material were studied to determine the effects of heat treatment on the size and dispersion of the UAl_2 and U_6Fe compounds. Samples were examined after anneals of up to 500 hours at 400, 450, and 500 C for specimens previously quenched from the beta and gamma phases. The U_6Fe phase shows pronounced growth even at the lower temperature. Precipitation of UAl_2 appears random in beta quenched material, but is grain or sub-grain boundary oriented in gamma quenched specimens after long alpha phase anneals. Specimens slowly cooled from the gamma phase developed a similar but more pronounced sub-grain network. This dispersion was not destroyed by beta heat treatment. This treatment provides a means of obtaining a markedly different dispersion of compounds than is obtained by only beta heat treatment of alpha worked material.

Uranium - Billet Fabrication Variables

Zircaloy-2 cladding thickness variation occurring during coextrusion is related to local variation in resistance to deformation of the uranium. This is believed to be a residual effect from the oriented large grain cast ingot structure. Development programs for evaluation of the effects of ingot structure on cladding variability include: (1) preparation of ingots containing different amounts of silicon for grain refinement of the cast structure, (2) testing the effect of heat treating billets in different ways, (3) increasing the amount of primary extrusion reduction by increasing the cast ingot diameter, and (4) combinations of the above. The first group of test materials received and coextruded as part of this program consisted of outer tube billets of standard composition, beta heat treated once after finish machining of the ID. Comparison of these test billets with billets heat treated before finish machining and with billets not heat treated showed some improvement in inner cladding thickness uniformity but no major differences as seen in Table I. The process route for production fabrication has not been changed. The billets are used in the as-extruded metallurgical condition.

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TABLE I.

Percent of Fuel Sections
with
Inner Cladding Thickness Variation
in
Excess of 0.008".

<u>Billets Heat Treated</u> <u>on Finished ID</u>	<u>Billets Not</u> <u>Heat Treated</u>	<u>Billets Heat Treated</u> <u>Before Machining</u>
7.7%	9.3%	12.4%

Zircaloy-2 - High Bend Ductility Strip Fabrication

Zircaloy-2 strip having uniquely high bend ductility, not available commercially, is required for fuel support fabrication. Development work has shown that proper crystallographic texture is required and may be produced by extrusion (either sheet bar or split tubing). The desirable orientation of the extruded stock may be retained through some reduction by warm and cold-rolling or turks head reduction.

Alternate End Closure Development

Gas Pressure Bonded Closure

Bonding of the Zircaloy-2 cap to the uranium core has been successfully demonstrated by the gas pressure bonding process. It has not been possible to satisfactorily bond the Zircaloy-2 cap to the Zircaloy-2 cladding with pressures up to 15000 psi and temperatures up to 650 C unless "diffuser" metals are used. Copper and titanium have shown favorable characteristics as "diffuser" materials. Development work is required to obtain continuous layers of these metals at the optimum thickness of 0.0005 inch.

Hot Headed Closure Studies

Recent effort on the Hot Headed Projection Welded and Bonded end closure was applied to the improvement of techniques in making a four ring projection type closure. Improved hot heading techniques provided elements with thicker cladding over the ends, permitting machining to eliminate the radius on the OD and ID. Welds made on the improved hot headed element with relocated ring projections show that the previous joint crevice condition has been virtually eliminated. Immersion plating procedures that

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are used to preplace thin films of copper in the areas between ring projections on the end caps have been modified; and as a result, the projection weld quality has been improved.

Metallographic examination of recent hot headed closures using a cap with four ring projections indicate that the projection welds are sound and the interface areas between the projections are bonded. Cross sections of the joint show that approximately 40-45 per cent of the joint is composed of projection welds, 18-22 per cent of the joint is the uranium-Zircaloy bond, and the remaining 33-42 per cent represents the Zircaloy-copper-Zircaloy bonds.

Self Brazed Closure

Three specimens, one of which had been machined at the closure end to lay bare the annulus of Ni-Cu-Zr braze between cap and clad, have been exposed for 85 days in the TF7 High Temperature Loop in K-East area. At the latest examination they showed no evidence of attack. Preparation of two lots of specimens for irradiation testing is in progress. One lot was beta heat treated before closure; the other was heat treated afterward. The objective is to verify satisfactory irradiation behavior of pre-heat treated fuel elements.

Irradiation Performance

KER Loop Irradiations

The post-irradiation evaluation of 14 NAE's ("N" reactor fuel elements) which were successfully irradiated at near prototypic N-reactor conditions to approximately 1200 MWD/T average exposure is continuing. The swelling behavior of these elements was described in the previous report. Study of pre- and post-irradiation dimensional data has revealed the behavior indicated below:

	<u>Inner Component</u> (NIE)	<u>Outer Component</u> (NOE)
<u>OD Increase</u>		
Range	0.003 to 0.007"	0.000 to 0.004"
Average	0.005"	0.002"

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	<u>Inner Component</u>	<u>Outer Component</u>
<u>Residual Warp</u>		
Range, Single Throw	0.005 to 0.025"	0.007 to 0.029"
Average, Single Throw	0.014"	0.012"
<u>Warp Change</u>		
Range, Single Throw	-0.007 to +0.015"	-0.001 to +0.016"
Average, Single Throw	+0.007"	+0.006"

The maximum calculated clad strain of the outer circumference of the inner component based upon the maximum observed OD increase is 0.56%, and of the outer circumference of the outer component is 0.16%. The planes of maximum warp in both sets of components were observed to shift in a random fashion.

Two outer components (NOE's) have been sectioned and examined in the Radiometallurgy facility. Fuel element performance appears to have been good. The Be-Zr eutectic brazed closures show no irradiation-induced alterations and no deficiencies are apparent. A slight buildup of corrosion product has been observed in a tight crevice between a spot-welded support and the fuel clad. No appreciable hydriding was observed in conjunction with the corrosion product. The nature of the deposit suggests that the corrosion occurred during autoclave testing and not during reactor irradiation.

The use of an alternative fuel other than the currently specified iron (150 ppm) and silicon (100 ppm) bearing uranium must be considered. U-2 w/o Zr alloy is a possible alternate fuel. "N" inner tubes with and without a 2 w/o Zr addition have been fabricated and prepared into test elements (NIN's). These alloyed and unalloyed elements will be irradiated as a single tube charge in the KER facilities for comparison of swelling behavior. Post-irradiation evaluation will include defect testing (simulated rupture) in the IRP facility.

KSE-5 fuel elements have been prepared and a charge of eight 12-inch long pieces is currently being irradiated to a goal of 1500 MWD/T. These elements are slightly modified KSE-3's (see previous reports) of 1.6% enrichment and are intended to operate at higher fuel temperatures so that additional swelling data may be obtained. KSE-3's operated at volume mean fuel temperatures in the range of 420-470 C, and KSE-5's will extend this temperature range to 520 C. The elements are currently operating at a maximum 190 kw/foot with heat fluxes in excess of 1×10^6 BTU/hr-sq.ft.

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Shaped Single Tube Element

An irradiation test to evaluate the fluted single-tube concept for N-reactor fuel is under way in the M-3 high pressure loop facility of the ETR. The test elements have eight flutes 0.1 inch high in the outer surface of the Zircaloy-2 clad uranium tube. Two elements have completed two reactor cycles for an exposure of about 400 MWD/T. They operate with a maximum metal temperature of about 520 C. The elements were examined after the first cycle and were found to be in excellent condition with no visible corrosion effects or distortion.

Post-Irradiation Non-Destructive Test Equipment

A non-destructive tester for irradiated N-fuel elements has been developed and installed in the 105 KE basin for final debugging using irradiated fuels. The unit is designed to measure: (1) internal and external clad to core bonding, (2) internal and external cladding thickness, (3) cap to core bonding, (4) cap to clad bonding, and (5) core integrity (cracks or voids in the fuel core material).

CURRENT REACTOR FUEL DEVELOPMENT

Alternate Process Development

The evaluation of an alternate manufacturing process for the production fuel elements for the Hanford production reactors exclusive of the NPR is continuing. A study of the technology, economics and manufacturing feasibility indicates that the hot-die sizing process has the most incentive. The main development effort has, therefore, been directed to evaluating the hot-die sizing process.

Hot Die Sizing

Hot die sizing has been developed into a two step process: (1) the die sizing step in which the lateral internal and external bond is formed, and (2) the end bonding step in which the end cap and base are bonded by a hot pressing step. The necessary equipment for bonding fuel elements by the hot-die sizing process became operational at HAP0 in the third quarter of CY 1962. A series of four variable analysis tests have been completed.

The first test defined the process variables for the die sizing step.

Preheat Furnace Temperature	680 + 15 C
Preheat Time ⁽¹⁾	12 + 1/2 minute
Sizing Speed	75 + 5 ipm
Reduction Ratio ⁽²⁾	45%

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- (1) The preheat time was increased from the previously reported 10-1/2 minutes because of changes in the furnace operating technique. The fuel assembly temperature remained the same.
- (2) The 45% reduction ratio is the current practice, but will be increased to 55% as soon as components are available.

Several subsequent tests and small pilot plant bonding runs have been made with this process and reproducible results were consistently obtained. A series of three tests have been completed in a program directed to defining the end bonding step in the hot-die sizing process. The results were previously reported from the initial two end bonding tests. The major problems encountered were primarily associated with end bonding containment die designs. A third test has now been completed with dies based upon the differential expansion concept.* The following process conditions were determined to be within a range that produced good end bonding and a variables test was made using all possible combinations of the following conditions:

Pressure	3.0, 4.0, 5.0 Tons/in. ²
Die Temperature	610, 640, 670 C
Pressing Time	2.5, 3.5, 4.5 minutes

*Sized pieces fitted with end bonding dies are preheated to approximately 400 C by heating for 5 minutes in a 680 C furnace.

The non-destructive testing data analysis has been completed and the conclusion was that no interaction between or within the range of variables existed. The individual fuel elements were of good quality. The destructive testing evaluation has been completed, but the statistical analysis of the data has not been completed. The results of the destructive testing has also shown that reactor grade fuel elements can be produced within the range of the test conditions.

Tentatively the following end bonding process parameters have been established:

Die Temperature*	640 $\pm$ 10 C
Pressure	4 $\pm$.25 Tons/in. ²
Time	4 minutes $\pm$ 10 seconds

*Use supplemental preheat 5 minutes in a 680 C furnace.

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The initial reactor production test fabrication has started. A tentative goal to fabricate a minimum of 200 fuel elements has been established.

Sylcor Hot Press

An initial shipment (320 pieces) of C4E enriched 0.94% hot press fuel elements fabricated by Sylcor was received at HAP0 in March, 1963. This is the first part of an order for 1000 fuel elements to be used for in-reactor evaluation comparison to ALSi bonded fuels.

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