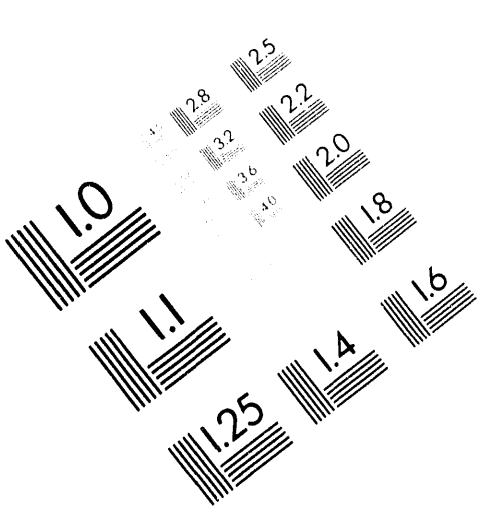

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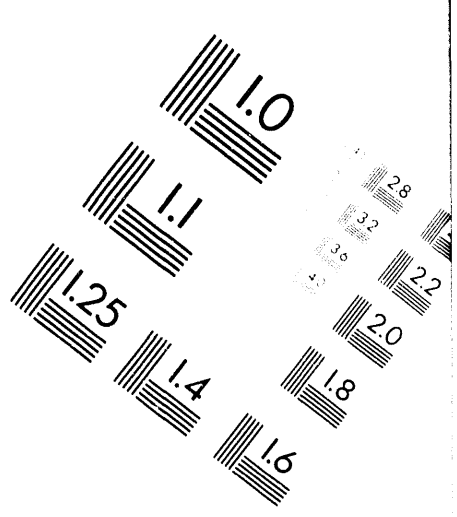

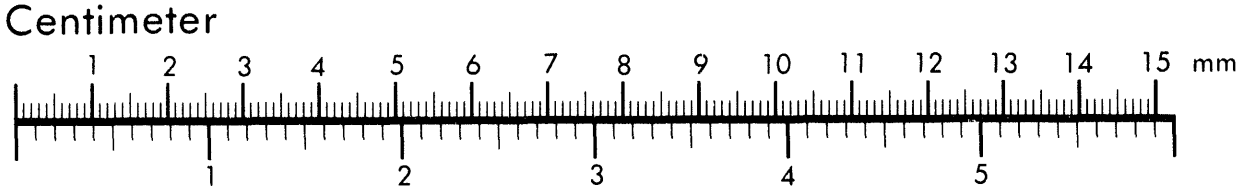

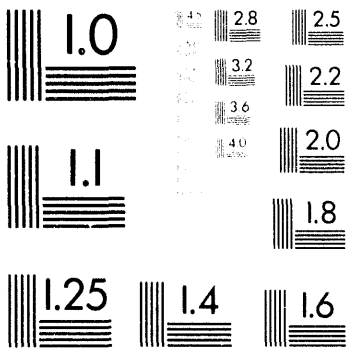

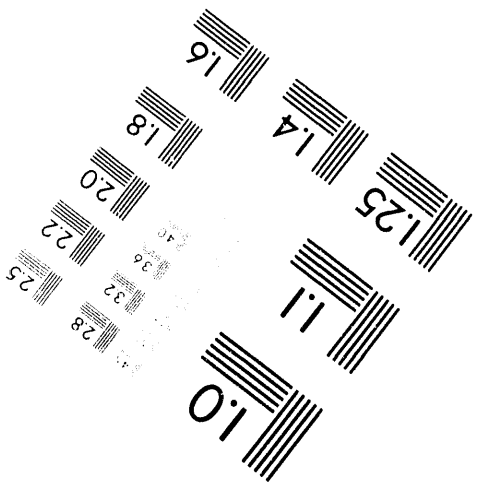

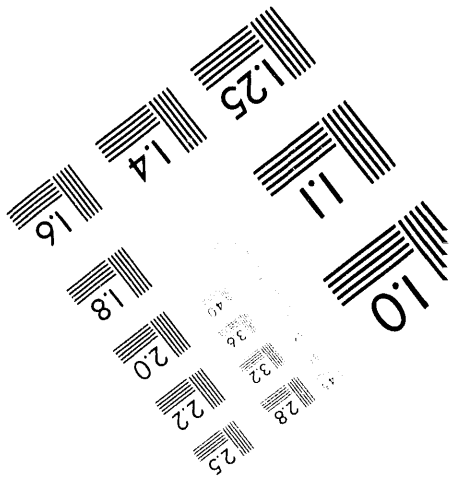

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REPORT TO THE WORKING COMMITTEE OF THE FUEL ELEMENT DEVELOPMENT COMMITTEE FROM THE GENERAL ELECTRIC COMPANY - HANFORD

by

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September 9, 1963

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TABLE OF CONTENTS

PRESENT REACTOR FUEL PRODUCTION

Uranium Core Production Data

Quality of Uranium Cores

Visual Inspection of Virgin Cores

Visual reject rates for the period June 1, 1963 to September 1, 1963 were 0.15 per cent for normal material and 0.3 per cent for enriched. The normal material has not changed significantly over the last year, while the enriched appears to have increased slightly during the reporting period.

UT-2 Grain Size Rejects

The UT-2 reject rate for the quarter ending September 1, 1963 was 0.4 per cent for normal ingot cores and 2.1 per cent for enriched cores. Dingot material averaged 6.2 per cent.

The decrease in the reject rate of normal cores from 1.4 per cent over the previous twelve months is due to more reliable operation of the UT-2 testers and increased uniformity of the tester standards. It is anticipated that efforts presently in effect will decrease the enriched rate a comparable amount in the next quarter.

Uranium Grain Size

The average grain size for cores received during the period March 1, 1963 to June 1, 1963 was 5.0, and no difference between enriched and normal was detected.

Hydrogen Analysis

The average hydrogen analysis on cores received during the period March 1, 1963 to June 1, 1963 was 1.2 ppm. No difference was detected between enriched and normal cores.

Braze Porosity in Normal Elements

Starting with cores shipped from NLO on May 1, 1963, a high incidence of external bond defects was detected in the canning of normal uranium. These defects appeared as voids in the external braze, and at times the UE-1 external bond tester rejected up to 15 per cent of the normal cores canned. Investigation of this problem showed that the rejects were highly ingot oriented, with some ingots showing as much as a 75 per cent reject rate. Also, it was observed that in all cases the high reject rate could be traced to ingots heat treated in the Amsler-Morton heat treating furnace between April 22, 1963 and June 7, 1963. The salt in the furnace was renewed on June 9, 1963, and no high reject ingots have been observed since that time. Studies at NLO indicate that the problem was probably caused

by a high hydrogen content in the "old" salt, and it is felt that it can be eliminated by more frequent cleaning of the Amsler-Morton furnace.

At the present time, two important questions are unanswered regarding this phenomenon. The first is an understanding of why certain ingots tend to show a high degree of porosity, while others heat treated under identical conditions show little or none. The second relates to the amount of hydrogen--either total or surface--which can safely be tolerated.

Oil Quenched Cores

Work is continuing in an effort to determine the feasibility of converting from water to oil quenching during beta heat treatment. All results to date favor oil quenching, and a reactor test on enriched cores is being run to duplicate the original normal uranium test before arriving at a decision. The recent braze porosity problem (see above) is additional incentive to convert to oil quenching, since the hydrogen pick-up is lower in this process.

Induction Heat Treating

Along with oil quenching, NLO has developed a method of induction heating the core blanks instead of heating them in a salt bath as presently done. From a fuel standpoint, the advantages of the process are lower hydrogen and less heat treat induced warp. An irradiation test is presently being run to determine if this material is dimensionally more stable during irradiation. If these results are promising, the possibility exists that a decision could be made regarding induction heating at the same time the final report on oil quenching is issued (January 1964). NLO has indicated that it would be economically desirable to make both conversions simultaneously.

Fuel Performance

Rupture Summary

Seven (7) natural and nine (9) enriched standard production fuel elements failed in the Hanford reactors during the period from April 1, 1963 through August 15, 1963. Irradiation and rupture classification data for these ruptures is summarized in Table I.

Fuel Performance Trends

The observed downward trend in fuel ruptures this period is attributed primarily to the virtual absence of cladding failures caused by non-uniform (groove or ledge) corrosion attack. During the first quarter of CY 1963, eighteen ruptures, seventeen in bumper fuel and one in non-bumper fuel, were due to this cause. Since that time only one rupture, a high exposure natural bumper failure incurred in April, has been attributed to groove or ledge corrosion. This reduction is believed to

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TABLE I

PRODUCTION RUPTURES

I&E Fuel Elements

Fuel Element Type	Reactor	Tube Power	Exposure MWD/T	Exposure % Goal	Rupture Classification	Failure Date
Enriched Non-Bumper	B	1168	485	70	Side Hot Spot	4/1/63
Natural Bumper	D	925	1062	99	Side Other	4/14/63
Natural Non-Bumper	B	1201	551	103	Unknown	4/16/63
Enriched Non-Bumper	B	1183	699	85	Side Other	4/20/63
Enriched Non-Bumper	C	1417	709	121	Side Hot Spot	5/7/63
Enriched Bumper	DR	1129	162	17	Side Mechanical Damage	5/10/63
Natural Non-Bumper	B	1130	46	8	End Mechanical Damage	5/22/63
Enriched Non-Bumper	B	1148	544	77	Unknown - Not Examined	6/6/63
Enriched Non-Bumper	C	1323	742	111	Side Hot Spot	6/18/63
Enriched Non-Bumper	B	1162	541	67	Side Hot Spot	6/20/63
Natural Bumper	DR	1164	667	83	Side Unclassified	6/24/63
Natural Bumper	D	1142	16	2	Side Mechanical Damage	7/1/63
Natural Bumper	D	1085	843	91	Side Unclassified	7/12/63
Enriched Non-Bumper	KE	1694	503	74	Side Hot Spot	7/22/63
Enriched Non-Bumper	KW	1711	801	109	Side Unclassified	7/30/63
Natural Non-Bumper	KE	1558	574	98	Side Parallel Split	8/5/63

[redacted]

have resulted largely from differences in reactor cooling water chemistry (pH reduced from 7.0 to 6.6), process tube age and condition. In addition, flow resistance was decreased in bumper fuel columns through the conversion from a rectangular solid rail to an elliptical bumper rail design.

With respect to fuel failures other than those caused by the above corrosion mechanism and mechanical damage, fuel performance was relatively unchanged.

AlSi Process Development

During this reporting period, AlSi process studies were primarily directed toward improving the characteristics of the AlSi-uranium bond. Major process developments were:

1. The nitric acid pickle time and duplex furnace parameters were optimized for the canning of the remaining dingot uranium cores in inventory. These changes were made to produce the less frangible UAl_3-type compound layers more nearly approaching ingot fuel quality. A high internal fractured bond reject was experienced. Further development revealed the action of the agitator to be at fault. Control of the AlSi freezeout on the core as it enters the lead is very critical as this governs the type, rate, and structure of the compound layer.

2. Studies of the interactions of process parameters on bond quality and closure integrity also led to further process optimization, and the application of more stringent inspection and testing standards to improve the quality and performance of enriched ingot fuel elements for the E-N loads. This technology is currently being applied to improve the quality of all fuel models.

3. Techniques were developed for applying a nonwetting agent (silicate solution) to the outer lip of aluminum cans to minimize AlSi slopover rejects. These rejects result from molten AlSi being drawn down between the can and sleeve by capillary action during the submerge portion of the fuel assembly operation. This process has significantly reduced the reject rate. A chemical milling process has been developed to reclaim the elements which are rejected.

Projection Fuel Program

A surface lubrication technique using two per cent soap solution for ultrasonically attaching the weld support to fuel elements has been improved by changing the lubricant to a light, high quality, mineral oil. This results in equal weld strengths, less weld time, and less operating and control problems. Full scale use of the oil lubricant is anticipated.

[redacted]

N-RD PRODUCTION FUELS

Fuel Production Status

The present production plan for the fabrication of N-Reactor fuel calls for a full charge of 372 tons of fuel to be ready for a November 15, 1963 fuel loading. The initial quantity of material to be available consists of about 400 tons of 0.947 per cent U-235 enrichment; a part of this tonnage is in a throwaway category. Some 1.25 per cent enriched material and some natural metal will be required for startup; the scheduling of these materials is not now complete.

To date, approximately 83 per cent of the load has been coextruded and 68 per cent has been autoclaved. Approximately 31 per cent of the fuel elements are completed and stored in shipping boxes. The backlog at support welding, reported in April, is being rapidly diminished.

The shop yield has increased to 81.5 per cent for July 1963 and the over-all shop yield is 68.4 per cent for the first six months of 1963. Cumulative rates of major reject categories for the first seven months of 1963 are shown below.

TABLE II

N-Reactor Fuel Reject Rates
First Seven Months CY 1963

Station	Reject Rate - Per Cent	
	Outer Fuel	Inner Fuel
Bonding (Peel Test)	2.6	6.6
Clad Thickness Variation	9.1	0.3
Warp	0.1	3.6
End Closure	11.3	4.0
Autoclave (No. of Failures)	2.0	1.0

The two autoclave failures shown for outer tubes in Table 2 were reported previously. They were attributed to clad penetrations during welding of the locking clips. The failure of the inner tube was caused by uranium contamination in the braze closure and was attributed to overheating of the uranium during brazing.

Uranium Composition: Phosphorus Contamination in Ingot Metal

During February 1963, phosphorus contamination in NPR ingot metal was detected and reported by the National Lead Company of Ohio. During March, Hanford set a temporary limit of 80 ppm phosphorus on all NPR billets. At that time, it was decided to coextrude all billets with less than 50 ppm phosphorus, while billets containing 51 to 80 ppm phosphorus were held pending evaluation for coextrusion performance and general fuel element fabricability. These billets were subsequently coextruded after successful fabricability evaluation.

[redacted]

The evaluation showed (1) the formation of the compound, UP, (2) a eutectic reaction between uranium and UP, and (3) the absence of solid solubility of phosphorus. Fuel elements were processed that contained up to 300 ppm phosphorus with no detrimental effects noted in either the fabrication or structure. Room temperature ductility was decreased as a result of phosphorus additions, but the effect was not apparent above 100 C. Twenty-one billets containing over 80 ppm phosphorus had been coextruded before phosphorus was detected. Approximately 28 outer and 180 inner acceptable fuel elements were made from these billets. These have not been removed from the first fuel supply. A study of fabrication yield on these billets disclosed no unusual reject rates nor have metallographic studies revealed any detrimental characteristics. Seven inner fuels, containing 168 ppm phosphorus, are being irradiation tested in KER 4 and are scheduled for discharge in November 1963. The exposure, as of August 16, 1963, was 592 MWD/T.

Bonding Problems

Unbond problems, which have been periodically experienced since November 1962, have been studied. Test work has resulted in minimizing the problem, but no clear-cut solution has yet been found. In addition to the improvement resulting from using thicker copper end plates, it has been noted that some billets were not thoroughly rinsed after cleaning on the I.D. Litmus paper tests on the I.D. of billets after final rinse showed several cases of acid solution remaining on the surface. Improved rinse methods, including a pressure I.D. rinse, have reduced this problem.

Current studies directed toward improved bonding include the studies of copper weld failures just prior to upset, a correlation of billet assembly component variables, and an improved flare or flanged can design to minimize tensile strain on the weld.

Braze Closure: Bevelled End Cap

The use of an end cap with a convex end surface (bevelled) rather than flat end surface, has been studied to determine if better braze metal flow and filling can be attained. In a test of 364 outer fuels, half of which were of the new cap design and half of the conventional design, a large decrease in defect rates was noted. The following table includes the results of the test:

TABLE III

Brazing Evaluation of Flat (Standard) vs. Bevelled End Caps

Type	No. of Ends Brazed	Braze Station	Bond Test	Braze Void	Weld	Total Defects	Per Cent Defective
Bevelled Cap	364	12	1	2	7	22	6.0
Standard Cap	348	11	7	13	13	44	12.6

Further evaluation is planned and a large group of bevelled caps are being procured for production testing of the new design.

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Materials

Uranium - Alternate Composition

Work is in progress to determine the effects of uranium alloy fuel compositions upon (1) fuel element fabrication, (2) irradiation swelling resistance, and (3) core corrosion in the event of a rupture.

The use of an increased amount of alloy content has been shown to have a beneficial effect in minimizing clad thickness variation (e.g., 2 W/o Zr-U and 800 ppm Al plus 400 ppm Fe). Use of a 2 W/o Zr-U alloy has reduced the rate of corrosion attack by the coolant on irradiated samples that have been purposely defected and tested in recirculating hot water loop facilities.

The beneficial use of alloys to minimize swelling has not yet been demonstrated. The 2 W/o Zr-U alloy in the stable two-phase state has shown greater swelling than has unalloyed uranium. The program on future alloy studies will have a three-fold approach:

(1) Investigate the use of Fe, Al, Si, etc., to form a finely dispersed second-phase precipitate (10^{15} particles per cm^3) in the uranium and serve as nucleating sites for fine porosity (i.e., small gas bubbles) in which surface tension is a major restraining force.

(2) Investigate the use of C, P, etc., to form a relatively coarser second-phase precipitate (10^{10} particles per cm^3) in the uranium. This work is being continued because of the observations made in post-irradiation annealing studies; a "halo" of gas porosity immediately surrounding uranium monocarbide inclusions is much finer than the porosity a few microns removed from the carbides.

(3) Investigate the concept of a "hybrid" fuel element composition having improved corrosion resistance (say Zr or Nb additions) and improved swelling resistance from either fine particles (say Fe, Al, Si additions) and for the "halo" effect of a somewhat coarser but more stable compound particles (say C, P additions).

Work has started with Fe and Al additions according to the first principle above and Al additions to the 2 W/o Zr-U alloy to study the "hybrid" concept.

Alternate Closure Development

Hot Headed Closure Studies

Development effort on the hot-headed projection welded and bonded closure has been applied to optimize various conditions of the welding procedures. These studies include: projection size, post heat conditions, variation of copper at the joint interface, and control of the O.D. dimension during welding. This work is continuing and test quantities of fuel elements will be prepared for nondestructive evaluation and irradiation testing.

End Closure Improvements - Beryllium Zircaloy Braze Ring Fabrication

Work has been initiated on improved braze ring manufacture. Samples have been prepared of rings formed from pressed, powdered, beryllium-zircaloy alloy. The test showed that this material fused below the specified maximum temperature of 985 C and the molten metal flowed smoothly. The melting time, however, was almost ten times that required for cast braze rings and is apparently due to the high resistivity of the oxide coated powder particles. Comparable melting times for cast, pressed chip, and compacted powder rings is 35 seconds, 90 seconds, and 330 seconds, respectively. Some improvement is expected through the addition of chips or wires to the powder rings by reduction of the resistivity of the conductive circuit.

The major benefits of the pressed powder braze ring are high materials utilization and decreased costs. Further work to improve melting time is in progress.

N-FUEL DEVELOPMENT

Pressure Bonded End Closure

The gas pressure bonded end closure for NPR fuel elements was developed to permit closure in the alpha temperature range of uranium. The technology of the gas pressure bonded end closure is capable of making a closure, with the weld removed, that will withstand a 750 F, 72 hour autoclave test.

Minimum bonding parameters are as follows:

Cap Type	Pressure psi	Temperature °F	Time Min.
Zr-2	30,000	1382	10
Zr-3	30,000	1202	10

It has been found that by a slight change in composition of the closure ring or by coating the ring with another metal (less than .001 inch thick) complete bonding of the Zr-2 closure ring can be effected below 1202 F.

Metallographic examination of the bonding has shown it to be complete and free of voids.

Self-Brazed Closure Development

A fuel element with closures fabricated by the "self-brazing" process was machined at one end to expose the braze layer around the entire periphery of the closure. The element was then autoclaved 72 hours in 300 C steam, with no apparent attack. It was transferred to the TF7 loop in the K-East test facility where it was subjected to 108 days exposure in 530 F water at pH 10. The braze layer appeared unaffected at the end of that period.

Two groups of elements, representing material that has been beta heat treated before and after closure installation, have been prepared for in-reactor testing.

Brazing and Braze Alloy Development

Corrosion Testing

Corrosion tests in 360 C pH7 water have exceeded 1300 hours on zircaloy-2 plus beryllium brazing alloy coupons, 1200 hours on zircaloy-2 plus uranium, and 450 hours on coupons of zircaloy-2 plus beryllium plus uranium. Comparative tests in pH10 360 C water have exceeded 400 hours. The following generalizations can be made:

1. Five per cent Be plus zircaloy-2 (standard braze) has a weight gain six times that of zircaloy-2 at 1300 hours and the color steadily becomes whiter indicating breakaway has occurred. Breakaway appeared to start around 700 hours.

2. After 1200 hours, zircaloy-2 plus 1/2 per cent U has essentially the same weight gain as zircaloy-2; zircaloy-2 plus one per cent U has a weight gain about two times that of zircaloy-2. The colors still remain glossy black.

3. The zircaloy-2 plus Be plus U series shows very little difference in weight gain with increasing uranium content; however, the weight gain increase is proportional to the beryllium content. With uranium present, breakaway corrosion occurs at less than 100 hours. The five per cent Be plus zircaloy-2 plus 2500 ppm U had a weight gain about ten times that of zircaloy-2.

4. An increase in pH from seven to ten has caused a 250 per cent increase in weight gains and a proportionally lower breakaway limit. The corrosion product on the five per cent Be alloys also appears to be less tenacious. The effect of uranium on the beryllium containing alloys was less in pH ten than pH seven. Zircaloy plus 1/2 per cent uranium had essentially the same corrosion rate as zircaloy-2.

5. Oxygen and nitrogen additions to the five per cent Be plus zircaloy-2 alloy resulted in very high weight gains and breakaway corrosion in less than 100 hours. Weight gains were in excess of 12 times those of zircaloy-2.

These corrosion tests will be expanded to determine the effect of iron, tin, and carbon on the properties of zircaloy-2 plus five per cent Be and to determine if the braze alloy can be desensitized to uranium contamination. The present tests will continue until the coupons disintegrate or show signs of complete failure.

Mechanical Properties

An investment casting process has been developed to make tensile and impact bars of the brazing alloy for testing. A wax pattern is first cast to the desired shape, an investment of suitable ceramic is made around the wax

pattern, the investment is cured, the wax melted out, then the mold is fired to remove the volatile constituents. The braze alloy is melted in graphite crucibles, then cast into the hot investment mold. After cooling, the mold is broken away and the specimen is ground to final dimensions. Excellent details have been cast including threads on the tensile bar. The only results to date are preliminary impact results which show impact strength of 3.31 cm - Kg/cm^2 (18.5 inch pound/$in.^2$).

Support Fabrication Development

In the assembly of the inner and outer tubes which make up the N-fuel, it is necessary to "size up" the inner supports to center the inner fuel and produce a specified spring constant in the inner fuel support system. This operation, which is done after the fuel components have been autoclaved, was producing cracks in some of the supports. The susceptibility to cracking was found to result from the presence of preferentially oriented hydride platelets which form in the zircaloy-2 during slow cooling in the autoclave. This trouble has been circumvented by annealing the autoclaved inner fuel at 300 C to solution treat the hydrogen in the supports followed by rapid cooling to reprecipitate the hydrides as finely dispersed particles which do not impair the ductility of zircaloy-2.

An alternate solution appears to exist in annealing the supports after fabrication. The zircaloy-2 strip for inner supports is highly textured to achieve the bend ductility necessary for forming the part; however, this texturing appears to promote the formation of large hydride platelets oriented with the forming direction. An alpha anneal at about 700 C will alter the texturing of the strip so that subsequent hydride formation is in the form of small platelets at grain boundaries and with random orientation. The ductility of parts annealed at 700 C is not impaired by autoclaving. This treatment is being studied for inclusion in the fabrication process.

Irradiation Performance

KER Loop Irradiations

A test charge of 14 NAE's ("N" reactor fuel element assemblies) has been successfully irradiated to an average tube exposure of 1800 MWD/T in a KER loop. The first 2/3 of the irradiation was at "N" conditions and the last 1/3 was at reduced coolant temperatures. Fuel swelling data obtained by the comparison of pre and post-irradiation fuel component bulk densities show a range of 0.0 to 1.0 per cent ΔV in the fuel of the outer component (NOE).

These swelling data with the exposure of the individual components are given in Table 4. In the table are similar data for the previous "N" prototype test that attained an average tube exposure of 1250 MWD/T. Selected components will be transferred to Radiometallurgy for detailed examination and others will be subjected to simulated rupture testing in the IRP facility.

Post-irradiation Radiometallurgical examination of the 1250 MWD/T exposure charge has been completed. Three outer and two inner components were examined

TABLE 4

POSITION IN CHARGE, EXPOSURE, AND MEASURED % ΔV IN THE FUEL OF TWO CHARGES OF NAE's

A. Charge Exposure = 1800 MWD/T

Element Position	Exposure of Inner (NIE) MWD/T	Exposure of Inner (NIE) a/o B.U.	% ΔV in Inner	Exposure of Outer (NOE) MWD/T	Exposure of Outer (NOE) a/o B.U.	% ΔV in Outer
Upstream	470	0.054	0	590	0.068	0.31
2	880	0.101	0.19	1100	0.128	0.49
3	1240	0.144	0.52	1570	0.181	0.73
4	1490	0.173	0.62	1880	0.218	0.95
5	1780	0.205	0.71	2240	0.258	1.42
6	2230	0.258	0.96	2810	0.324	1.66
7	2440	0.282	0.80	3080	0.356	2.16
8	2460	0.284	0.76	3090	0.357	2.32
9	2420	0.279	0.76	3040	0.352	2.49
10	2050	0.237	0.65	2580	0.298	1.37
11	1520	0.176	0.40	1920	0.222	0.88
12	1120	0.129	0.31	1410	0.163	0.69
13	870	0.100	0.20	1100	0.126	0.56
Downstream	640	0.074	0.03	800	0.093	0.36

B. Charge Exposure = 1250 MWD/T

Element Position	Exposure of Inner (NIE) MWD/T	Exposure of Inner (NIE) a/o B.U.	% ΔV in Inner	Exposure Outer (NOE) MWD/T	Exposure Outer (NOE) a/o B.U.	% ΔV in Outer
Upstream	360	0.042	0.03	450	0.052	0.38
2	800	0.093	0.40	1010	0.116	0.54
3	1190	0.137	0.54	1490	0.173	0.62
4	1430	0.165	0.60	1800	0.208	0.69
5	1490	0.173	0.54	1870	0.217	0.72
6	1480	0.170	0.55	1850	0.215	0.90
7	1400	0.162	0.68	1760	0.204	0.92
8	1380	0.159	0.77	1730	0.200	0.77
9	1360	0.158	0.86	1710	0.198	1.08
10	1250	0.144	0.63	1570	0.181	0.93
11	1030	0.120	0.55	1300	0.150	0.58
12	740	0.086	0.43	930	0.108	0.50
13	640	0.074	0.29	800	0.093	0.45
Downstream	430	0.050	0.09	540	0.063	0.47

in detail. The components appeared to be free of any clad striations, bumps, or indications of clad thinning. The Be-Zr eutectic brazed closures for both components show no deficiencies in their irradiation performance. Crevices between the clad surfaces and the spot-welded support tabs have been examined and no indications of in-reactor corrosion or crud deposition have been observed. Zircaloy-2 samples of the clad and the supports have been analyzed for hydrogen with results in the range of 60-110 ppm. No localized or large concentrations of hydrides have been seen metallographically. An outer tube suitcase-handle type support was mechanically tested and comparison with pre-irradiation test data indicates that crushing strength and deformation characteristics have not been adversely affected by the in-reactor service. Contact areas between the inner surface of the outer tube and the inner tube supports have been examined with no evidence of fretting corrosion being observed.

A charge of KSE-5 single tube fuel elements has been discharged from a KER loop after attaining an average exposure of 1240 MWD/T at volume mean fuel temperatures ranging up to 535 C. The fuel swelling data obtained by comparison of the pre and post-irradiation fuel element bulk densities ranged from 0.78 to 2.28 per cent ΔV. These new data from the eight-twelve inch long elements extends the knowledge of the uranium swelling behavior to a temperature region 70 C above that obtained from earlier irradiation testing of KSE-3 elements. The operational data and the measured fuel swelling of the KSE-5 elements are tabulated in Table 5, along with similar data of 1200 exposure KSE-3's. The observed fuel swelling of the two test elements (KSE-3 and 5) appear consistent with each other and indicate that swelling begins at a mean fuel temperature around 400 C and approaches a theoretical maximum, which is established by the loop coolant pressure and fuel temperature, at a mean fuel temperature around 525 C. The uranium swelling model will be further evaluated using the new KSE-5 data (the swelling model has been previously discussed and describes fuel behavior of an incremental volume in terms of burnup, temperature, and restraint).

TABLE 5

Position in Charge, Exposure, Vol. Mean Fuel Temp., and Measured % ΔV in the Fuel of KSE-3's and KSE-5's Irradiated to ~ 1200 MWD/T

Element & Position	Exposure MWD/T	Exposure a/o B.U.	Vol. Mean Fuel Temp. C	% ΔV in Fuel
KSE-3 Upstream	1190	0.138	430	0.53
" 2	1210	0.140	450	1.35
" 3	1210	0.140	460	1.37
" Downstream	1170	0.135	465	1.32
KSE-5 Upstream	1070	0.123	433	0.78
" 2	1160	0.135	463	1.07
" 3	1260	0.145	495	1.72
" 4	1320	0.152	518	2.12
" 5	1350	0.156	534	2.28
" 6	1320	0.152	535	1.95
" 7	1290	0.149	535	1.97
" Downstream	1160	0.135	511	2.06

Tapered End Cap Design (The V-Bottomed End Cap)

N-inner fuel elements with tapered end caps (four braze-bonded and four unbonded) completed irradiation to 1900 MWD/T in high temperature water. The calculated surface temperature ranged from 168-226 C. The elements operated at specific power above that anticipated for N-reactor conditions. Density measurements in the basin indicate the swelling to be 0.5 - one per cent with no significant difference between the bonded and unbonded elements. Visual examination in both the basin and in air in radiometallurgy showed no abnormalities in any element. No evidence of the clad shearing mechanism was observed on either the tapered end cap designs and was experienced with flat, flush, unbonded end caps.

Shaped Single Tube Element

Two different fuel element geometries are included in studies to evaluate the irradiation performance of fluted fuels. One is an N-reactor single-tube element with eight flutes on the outer surface. The other is a six-fluted element which is approximately the size of the present N-inner fuel tube.

Two of the N-reactor single-tube size elements are being irradiated in the M-3 high pressure loop of the ETR. Operating at a power of about 130 kw/ft. with a maximum metal temperature of 520 C, these elements have accumulated an exposure of about 1000 MWD/T. Interim examinations of these elements have been made. There has been no visible change in the elements, and weight measurements indicate that about 0.3 per cent, or the theoretical minimum, fuel volume increase has occurred.

One of the small six-fluted elements has been irradiated to an exposure of 1200 MWD/T in high temperature water in the P-7 loop of the ETR. With this exposure, the element has undergone a volume increase of about three per cent. This test will continue with interim examinations until an exposure of about 3000 MWD/T is obtained.

The cold water irradiation of three of the small fluted elements was terminated at 2000 MWD/T exposure by the failure of one of the elements. The post-irradiation examination of these elements from the cold water irradiation has been started. Localized areas of pronounced swelling exist near the ends of these elements, and the failure of one element was prabably the result of this swelling. The post-irradiation examination has not proceeded far enough to determine the nature of the swelling.

CURRENT REACTOR FUEL DEVELOPMENT

Alternate Process Development

Fuel cladding processes investigated at Hanford as possible alternates for the present AlSi bonding process are as follows: (1) Sylcor hot press, (2) HAPO hot press, (3) fluid pressure bonding, and (4) hot die sizing. Each of these processes produce diffusion bonded fuel elements through the application of pressure and temperature to aluminum-clad, electro-nickel plated uranium cores.

Initial studies indicated that the hot die sizing process was the most attractive in terms of over-all economics, flexibility, and control. Consequently, major emphasis was placed on the development of this process for evaluation during FY 1963 and is continuing in FY 1964.

Hot Die Sizing

An initial in-reactor production test was fabricated during June and July. The fuel elements proved to be of reactor grade quality as determined by extensive destructive and nondestructive testing.

Twenty weighed and measured charges of normal fuel elements for in-reactor testing were charged at C-Reactor July 27 for irradiation to 800 MWD/T. These are presently at 50 per cent of goal. Three of the charges were thermocouple columns, two hot die size, and one AlSi bonded control charge. Seventeen columns were striped charges, each consisting of matching alternating hot die size and AlSi bonded fuel elements in the downstream half of the column; the remainder of the pieces in the seventeen columns were standard AlSi bonded fuel elements. A total of 200 hot die size and 408 AlSi bonded controls was charged in the irradiation test.

A second production test using (0.947 per cent) six-inch enriched cores has been fabricated and the necessary AlSi bonded control fuel elements fabricated. This production test is designed to evaluate the hot die size bond under more severe reactor operating conditions. Only slight modifications to the process established for the eight-inch normal fuel elements were required.

Processing Conditions

A. Sizing Six and Eight-Inch Models:

Temperature and Preheat Furnace:	680 ± 15 C
Time:	12 ± 1/2 minute
Sizing Speed:	65 ± 5 ipm
Reduction Ratio:	45 per cent

B. End Bonding:

	Eight-Inch Model	Six-Inch Model
Supplementary Preheat:	6 minutes	5 minutes
Bonding:	4-1/2 ± 1/8 minute	4-1/4 ± 1/8 minute
Pressure:	4 ± 1/4 ton/in.^{2}	4 ± 1/4 ton/in.^{2}
Head Temperature:	655 ± 10 C	655 ± 10 C

Complete provisional process specifications for the hot die sizing process have been issued in HW-77683. These specifications were developed for the CDB2N model and some slight process adjustments may be required for other model types.

A group of four titanium-tungsten carbide dies, two smooth bore, one four-rib slotted and one eight-rib slotted, for the hot die sizing process have

been used to produce fuel elements with integral ribs in lieu of ultrasonic welds for self-supported fuel elements. Initial results are encouraging. Integral ribbed cans with 60 mil high ribs have been produced from smooth wall 85 mil thick aluminum cans. No galling or tearing of the ribs was noted. Figure 1 shows a die sized fuel with integral ribs.

A review of the reactor performance of HAPO fuel elements points to the fact that non-uniform corrosion of the 8001 alloy cladding is a leading problem. This problem is even more evident when long residence times (nine months vs. 2-1/2 months) are considered for HAPO fuel elements. The problem is universal for the HAPO old reactor complex regardless of the bonding process. Programs have been initiated to investigate the possible use of high silicon alloys in lieu of 8001 alloy for the hot die size bonding process. There are several apparent incentives for this change:

1. The presence of silicon in the Ni-Al bond zone enhances the metallurgical structure of the bond by the suppression of the formation of Ni_2Al_3. Preliminary testing and metallography have repeatedly demonstrated this important fact.

2. The use of silicon alloys may permit the elimination of nickel from the cladding, thus enhancing the reactivity of the fuel element, or at least permit less nickel.

3. The use of ten or eleven per cent silicon in a basic 8001 alloy has, based on preliminary tests, given excellent uniform corrosion resistance and probably has very good non-uniform corrosion characteristics.

Sylcor Hot Press

A total of 781 C4E enriched (.94 per cent) hot press fuel elements fabricated by Sylcor under contract DDR-149 was received at HAPO in April 1963. Preliminary evaluation of these fuel elements revealed that the bonding was excellent. The closure welding, however, did not meet specifications. Pinholes in the welds and lack of coverage will require rewelding of all the fuel elements. Two attempts to define a welding process to weld these fuel elements effectively have not been successful.

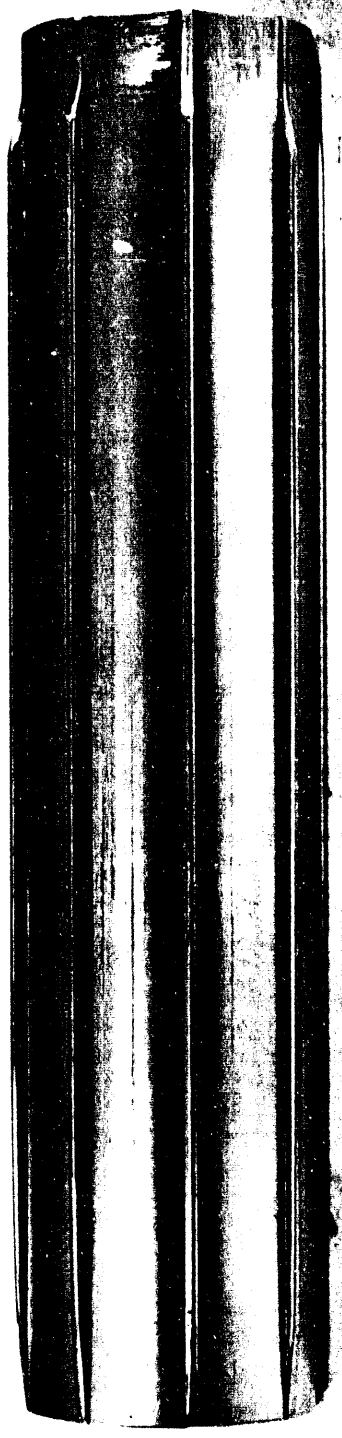

FIGURE 1

HOT DIE SIZED FUEL
ELEMENT WITH INTEGRAL RIBS

DATE
FILMED
7/5/94
END