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WORKING COMMITTEE REPORT FROM
THE GENERAL ELECTRIC COMPANY - HANFORD

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April 8, 1959

by

J. T. Stringer

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WORKING COMMITTEE REPORT FROM
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PRODUCTION REVIEW

1. URANIUM QUALITY AND TECHNOLOGY

1. Quality of Uranium Cores

During this reporting period, the metal quality level as measured by acceptance and in-process inspections decreased slightly from the level for the previous period. As shown in Table I , the average rejection rate for surface defects in virgin material is about 1.2 percent compared to less than 1 percent for the previous period.

The average rejection rate for surface defects in recovered material, as shown in Table II , has remained at the higher levels first experienced in October 1958. However, in February and March, there appears to be a decreasing trend developing. In attempting to correlate this increase in rejection with the metal history, it is interesting to note that a fairly direct correlation seems to exist between HAPO rejection rate and FMPC yield as shown in Figure I. In projecting data beyond that shown in Figure I , it appears that a decreasing trend in FMPC yield may have started in November and December 1958 as shown in the MQWC report for January 1959. With a three month delay period, this would correlate fairly well with the decreasing trend in HAPO rejection rate which appeared to start in February and March 1959. Whether this correlation will be shown to be significant is in question; however, in any event, the problem certainly requires serious consideration. Also, shown

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in Table II are the rejection rates for enriched metal cores which average about 5.8 percent as compared to 15.8 percent for normal metal cores.

As shown in Table III, the total porosity type rejects have fluctuated considerably during this period with most of the significant fluctuation being in the bond test category. Most of the periods of high rejection for bond test are for core types of which very small quantities were processed. The average void rejection rate continues to remain well below the one percent level.

Shown in Table IV is a comparison of the metal quality rejection rates for the major categories of chloride and carbonate heat treated material processed since the start of chloride heat treating in November 1958. For the core types of which significant quantities have been processed, the rejection for porosity-type defects continues to be lower for the chloride material. The Sonotest rejection rate for chloride, 19 sec. delay, is about 10.5 percent compared to 2.5 percent for carbonate. The rejection rate for approximately 10,000 pieces of 43 sec. delay processed to date is 3 percent. Added control of the heat treating process variables is expected to reduce this level below that of the carbonate material.

The downward trend in reactivity of core which started at the end of the last reporting period has subsided and a leveling off trend appears to be developing as shown in Figure II.

2. Fuel Core Casting

The current status of the fuel core casting program is essentially the same as for the last reporting period. FMPC is continuing its development work to eliminate the porosity and high carbon content problems encountered in the first run of cast cores received at HAPO in September 1958. To minimize and, if possible, eliminate the need for heat treating, an alloying program for cast cores is being studied by FMPC and HAPO. It is hoped that sufficient grain refining and satisfactory orientation can be obtained in the as-cast cores and thereby improve the quality and the economics of casting.

3. Alpha Annealing

The results of the test of improvement in reactor stability of alpha annealed cores are summarized in Table V. In analyzing these data for significance, the most important result was that the alpha annealed was significantly low with respect to tube filling capacity values (combination of warp and diameter changes). At low exposures, the effect was not strong, but increased exposure strengthened the effect such that at higher exposures, the difference was strongly significant. In view of the significance of this test, it is planned to place an order for 5,000 alpha annealed cores for large scale reactor testing.

4. Dingot Utilization Program

On the completion of the evaluation of the 65 tons of dingot cores

received in January 1959, it was concluded that the dingot was substandard in quality and was unacceptable for irradiation because of the following:

1. The grain size was too large and was in the range where bumping during irradiation has been experienced.
2. The crysrallographic orientation as measured by x-ray diffraction techniques and the Sonic Orientation Resonance Test was less desirable and less uniform than for ingot and was considered to be in the range which produces serious dimensional instability during irradiation.
3. The hydrogen content of the dingot was not distributed as expected, but was distributed toward the high side of the range of 0 - 5 ppm with some material exceeding 5 ppm. Because of fuel element performance considerations and fabrication economics associated with bond porosity, it is important that the hydrogen content be controlled at least to the range of ingot (2 ppm or less).
4. The metal soundness was found unsatisfactory because of the presence of extrusion defects in the finished cores. Because of the nature of the defects and the absence of means for detecting all the defects in the shipment with the necessary assurance, the entire shipment was rejected. Extrusion and/or inspection procedures must be developed to assure defect-free cores

Because of the importance of continuity of operation of the Weldon Spring Plant, and the attaining of an economic level of production, the dingot problem was thoroughly reviewed and the following two-fold utilization program was established.

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a. Dingot of lesser quality than ingot.

Specifications:

Grain Size: Being determined

Orientation: Being determined

Hydrogen Content: 5 ppm or less, heat treated in chloride salt.

Metal Soundness: Extrusion defect free.

Quantity:

Twenty-five tons per month for the first six months with the quantity increasing linearly to 100 tons per month, 18 months after initial acceptance. This utilization schedule is based on the assumption that no solution to the dingot problem is forthcoming in the first six months.

b. Dingot of quality comparable to ingot.

Specifications:

Grain Size: Being determined

Orientation: Being determined

Hydrogen Content: 2 ppm or less, heat treated in chloride salt.

Metal Soundness: Extrusion defect free.

Quantity:

Sixty tons per month for the first six months, 150 tons per month for the second six months, and acceptance thereafter based on experience obtained during the first year of dingot use.

5. Uranium Core Product Specifications

In establishing and maintaining a comprehensive quality control program for uranium cores, it is desirable, if possible, to establish for the producer sites product specifications for the various characteristics requiring control. Of course, inherent in such a program is the need for equipment and procedures for measuring these characteristics on a production basis. The following outlines HAPO's activities in the areas of specification definition, and equipment and procedures development for the control of grain size, orientation, internal defects, surface defects, and striation content.

a. Grain Size

The age old problem in specifying the grain size of uranium has been the absence of an adequate method for measuring that can be applied by all sites and result in acceptable reproducibility of measurements. In hopes of providing a uniform method, HAPO has been engaged in the development of a nondestructive testing instrument (UAT) for measuring average grain size. Although this instrument is still in the prototype stage, the evaluation results to date are most encouraging. In Figures II-A, II-B, and II-C are shown the grain size distributions of ingot and dingot as determined by the Ultrasonic Attenuation Test (UAT), the Heyn Intercept Method, and visual comparison of 3X macrographs. It should be noted that the sample sizes were not the same in all cases, but nevertheless, the distributions take on similar shapes and ranges and appear to indicate that

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UAT holds considerable promise as a satisfactory grain size measuring device. In the absence of an approved nondestructive test, the Heyn Intercept Method is considered to be a useful tool on a statistical basis.

b. Orientation

The problem in specifying the desired orientation of uranium cores has been the absence of a method readily applicable to production quantities. The evaluation results to date of the Sonic Orientation Resonance Test at HAPO indicate this instrument may provide a useful tool for the measurement of core stability. It is presently indicated that SORT is sensitive to both stress and orientation. Should this be proven to be the case, it is felt that SORT would still be a useful tool since both stress and orientation are considered to influence reactor stability of cores. In the absence of a nondestructive test, x-ray diffraction techniques and the Growth Index value are considered useful tools on a statistical basis. Shown in Figures II-d, II-e, II-f, and II-g are distributions of SORT values, Growth Index values, and dimensional changes and warp during irradiation for ingot and dingot. Considering SORT to relate stress and orientation level, it is interesting to note the similarity in the SORT and TFC distributions in Figures II-d and II-e, respectively. Considering Growth Index relates only orientation, it is also interesting to note the similarity in the G.I. and Diameter change distributions in Figures II-f and II-g, respectively. It should be noted that these distributions are

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based on limited samples and in some cases different populations.

However, it is felt that the trends indicate that there may be useful methods in either SORT or G.I. for predicting core stability.

c. Internal and Surface Defects and Striation Content

The development of equipment and procedures for controlling the quality from the standpoint of internal defects and surface defects is in the stage of determining the capability of current prototype models in adequately detecting such defects. Striation content control by nondestructive methods is only in the feasibility study stage.

d. Equipment Development and Application Schedule

Assuming that nondestructive testing technology progresses as anticipated and significant correlations are established between the tests and the characteristics being measured, the anticipated schedule for continued development and application of nondestructive test equipment is shown in Table VI.

e. Tentative Product Specifications

To relate to the producer sites what is anticipated in the way product specifications, tentative specifications for grain size and orientation are shown in Table VII. Tentative specifications are listed for current measuring methods and for nondestructive test equipment methods since the acceptability of the nondestructive equipment is not fully established. It should be emphasized that these are based on limited data and are listed only as comparative values for ingot and dingot and are not yet considered absolute

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values. It is hoped that with the collection of additional data such specifications can be firmly established.

B. CANNING EXPERIENCE AND PROCESS TECHNOLOGY

Solid fuel elements and the following types of I & E fuel elements were canned in the 313 Building during the last four months:

- 8" normal uranium C size; 40 mil can wall; 1.460" O.D.; 0.375" I.D.
- 8" normal uranium O size; 40 mil can wall; 1.445" O.D.; 0.310" I.D.
- 8" normal uranium KII size; 40 mil can wall; 1.474" O.D.; 0.375" I.D.
- 8" normal uranium KIII size; 40 mil can wall; 1.460" O.D.; 0.385" I.D.
- 6" 0.94% enriched uranium C size; 38 mil can wall; 1.454" O.D.; 0.375" I.D.
- 6" 0.94% enriched uranium O size; 38 mil can wall; 1.440" O.D.; 0.310" I.D.
- 6" 0.94% enriched uranium K size; 38 mil can wall; 1.454" O.D. 0.400" I.D.

Production quantities of the KIII normal fuel elements and the K size enriched fuel elements were canned for the first time in January and February respectively.

Quantities and yields of production fuel elements canned during the last four months were:

<u>Type of Fuel Element</u>		<u>Dec.</u>	<u>Jan.</u>	<u>Feb.</u>	<u>March</u>
8" normal C size	No. Canned	2,276	12,503	666	26,001
	% yield	64.8	82.1	76.5	83.6
8" normal O size	No Canned	80,885	57,309	76,818	69,100
	% yield	83.0	84.9	87.7	82.0
8" normal KII size	No Canned	67,685	45,616	15,478	35,133
	% yield	81.0	80.7	77.6	77.5
8" normal KIII size	No canned		12,950	4,038	23,344
	% yield		81.1	83.8	83.3

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<u>Type of Fuel Element</u>		<u>Dec.</u>	<u>Jan.</u>	<u>Feb.</u>	<u>March</u>
6" 0.94% enriched C size	No. Canned	20,838	20,425	13,453	1,054
	% yield	85.8	88.0	83.9	87.1
6" 0.94% enriched C size	No. Canned	1,674	7,512	6,221	6,902
	% yield	69.9	81.8	82.6	80.3
6" 0.94% enriched K size	No. Canned	--	--	8,949	10,325
	% yield	--	--	75.9	81.1
8" solid	No. Canned	1,527	8,693	24,148	21,685
	% yield	78.1	86.5	86.1	79.5
6" enriched solid	No. Canned	146	4,176	--	790
	% yield	100.0	78.6	--	82.5
Total No. Canned		175,031	169,184	149,711	196,031
Overall % yield		82.2	83.4	84.8	81.2

The average yield for I & E fuel elements for December through March was 82.9%. This is an increase of 5.5% from the August through November yield. The biggest reduction (2.9%) was made in penetration rejects. The four highest reject classifications for December - March were bond test (4.6%), marred surfaces (2.9%), assembly rejects (2.5%), and penetrations (2.3%). Some of the major changes made in the canning process were:

1. The sleeve life was reduced to twenty-five cycles in December. This is probably the main reason for the reduction (2.9%) in the penetration reject rate.
2. Bump discriminator probes were installed on all internal penetration testers in December.
3. Vibration of the canning jacks was improved in January by increasing the amplitude and frequency of vibration.

4. In February an improved tape head replaced the servo head on the penetrator testers.
5. Production use of the automatic welders was started in March.
6. Accepting internal penetration reject lots on the basis of destructive testing of a statistical sample was stopped in March.
7. Reheat-treating of transformation test rejects was stopped in March.

The biggest improvement in fuel element quality is expected from changes (6) and (7) above. These two groups of fuel elements have had a high percentage of the internal pile ruptures. Caustic etching of the cap wafer has reduced the number of autoclave failures from an average of two a month for the first eleven months of 1958 to one-half a month for December through March.

C. CANNING PROCESS DEVELOPMENT STUDIES

Production methods of attaching collapsible aluminum alloy supports to self-support fuel elements (shown in Figure III) and to bumper fuel elements are under development. The self-support fuel elements are used in ribless process tubes and the bumper fuel elements in the present ribbed process tubes. Bumper supports are not as high as self-support spacers.

The present method of attaching supports on production test quantities (about six tubes per month) are by spot welding them to the aluminum jacket.

Ultrasonic welding and a stud welding technique are also being investigated.

Temperature profiles at canning jack positions in a canning furnace have been found to be a method for optimizing furnace tap settings for different can-sleeve

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sizes to eliminate large temperature gradients. These profiles must be made for each individual canning furnace.

The present canning process has been found to operate in a border line zone which is conducive to Al-Si freeze out in the heavy can base during can submersion. If insufficient time is allowed for Al-Si to melt before inserting the cap-spire, void areas and non-wetting at the can base and spire wrinkling can occur. An optimum I & E canning cycle has been selected by statistical tests which has eliminated this bottom freeze out problem. This is a 45 second cycle with a supplemental can-sleeve assembly preheat position which will give 73 seconds can-sleeve preheat and 30 seconds submerge time when using standard 11.2% Si Al-Si.

Spire-crack rejects have been found in X-8001 and C-64 alloy cap-spikes and resulted in about a 0.3% reject rate during the first part of March. The spire⁵ crack during the canning operation just below the cap-wafer-spire-radius junction allowing Al-Si to enter the tube as shown in Figure IV and V. At this location it is now impossible to detect these cracks with the internal penetration tester and it is conceivable that some of the internal reactor failures could be due to spire-cracks. No examples of partial spire-cracking have been found to date and no fuel elements with spire cracks have been found past the weld and mandrel inspection station.

Cocking the cap wafer during spire-core seating or during quenching has been found to produce gross numbers of typical spire-crack rejects. Twisting, bending,

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or striking the spire during assembly has also been found to produce spire-cracks.

D. IRRADIATION EXPERIENCE

1. Rupture Summary

Fourteen normal uranium (13 I & E and 1 solid), two enriched (0.94% U-235) I & E, and three depleted I & E fuel elements failed in the reactors during the period from December 15, 1958 through March 31, 1959. A summary of these ruptures is as follows:

	Date	Reactor	Exposure (MWD/T)	Tube Power at Failure (KW)	Type of Fuel Element	Rupture Classifi- cation	Cladding Alloy
1958	12/27	C	733	691	Depleted I&E	Internal	1245
	12/28	KE	743	1068	Normal I&E	Internal	X-8001
1959	1/12	C	688	667	Depleted I&E	Internal	1245
	1/14	C	729	658	Depleted I&E	Internal	1245
	1/16	F	722	972	Normal Solid	Unclass.	C-64
	1/19	KW	809	995	Normal I&E	Internal	C-64
	1/28	KW	566	1050	Normal I&E	Internal	X-8001
	1/28	KW	686	1228	Normal I&E	Internal	X-8001
	2/4	KE	653	1234	Normal I&E	Internal	X-8001
	2/16	H	893	983	Enriched I&E	Side Hot Spot	C-64
	2/24	H	801	858	Enriched I&E	Internal	C-64
	2/28	KE	735	1272	Normal I&E	Side Hot Spot	X-8001
	2/28	KE	813	709	Normal I&E	Internal	X-8001
	3/7	DR	828	1010	Normal I&E	Internal	C-64
	3/13	D	993	876	Normal I&E	Internal	C-64
	3/17	DR	713	918	Normal I&E	Unclass.	X-8001
	3/24	KE	711	1274	Normal I&E	Side Other	C-64
	3/24	KW	711	1274	Normal I&E	Unclass.	X-8001
	3/31	B	--	1013	Normal I&E	Internal	X-8001

2. Rupture Trends and Experience

A more favorable trend in rupture frequency was established during this period.

Normal and enriched fuel element ruptures decreased 48% and 85%, respectively, from the number experienced during the previous 3-1/2 month period. This reduction is largely attributed to I & E loadings in the central tubes (75% of total) of all

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reactors, lower reactor operating temperatures, decreases in I & E variable goal exposures, and the probable effects of tighter bond test standards and process changes to improve fuel element bond quality.

As indicated above, internal ruptures are currently the major I & E rupture category. Radiometallurgical examination has led to the conclusion that the majority of these ruptures have resulted from coolant entry through the cap closures. Underlying non-wet and/or porous braze areas permitted coolant passage along the spire to a point where sufficient water-uranium reaction occurred to collapse the spire wall and in some cases fracture the uranium core. Thus far water entry through the cap closure has been definitely confirmed in the examination of three normal uranium and three depleted I & E internal ruptures. One enriched I & E internal failure showed evidence of intergranular corrosion of the spire, however, it is not known whether this condition caused the rupture or occurred as a result of coolant flow restrictions at the time of failure. Further examinations of internal ruptures are scheduled to determine if other mechanisms are involved.

I & E fuel elements clad in X-8001 alloy appear to be more prone to fail internally than those clad with C-64 alloy. Presumably this difference results from the X-8001 alloy being more difficult to wet with the Al-Si bonding medium. Recent process developments have shown this problem can be satisfactorily resolved.

Because of the high incidence of depleted I & E internal ruptures (13 total) in a test to produce high Pu-240 product, the remaining 43 columns of the original 84 columns were discharged in January, 1959, after reaching about 60 per cent of goal.

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Irradiation of 8 columns replaced with depleted I & E elements, prepared using a caustic etchant on the caps to improve closure integrity, has continued without incident on a run-to-rupture or maximum exposure time of one year.

External jacket corrosion ruptures have decreased as expected, considering the lower reactor operating temperatures. Forthcoming seasonal uses in reactor coolant inlet temperatures and planned increases in power levels will undoubtedly increase the frequency of jacket corrosion failures. Since fuel element warp is one of several factors that will cause flow restrictions in reactor process tubes and ultimately leads to severe fuel jacket corrosion, a recent incidence of eleven "O" size I & E charges requiring abnormal discharge pressures because of excessive fuel element warp is highly significant. Maximum warp of these charges ranged from 60 - 70 mils. One fuel element, which warped 60 mils, showed evidence of surface bumping 4 - 6 mils in height. This fuel element is scheduled to be examined in the Radiometallurgy Laboratory to determine if anomalies in uranium core structure contributed to instability and bumping. In addition ingot numbers have been obtained from the ends of the warped fuel cores to facilitate investigation of fabrication history

3. Present and Future Reactor Environments

Reactor power levels were increased 5 - 10 per cent, following the relaxation of the ACRS limits in January, 1959. Plans for further increases in power levels include operating all reactors on 93.5 C bulk outlet temperature limits by this summer and gradually increasing levels thereafter to attain 95 C bulk limits by FY 1962, providing fuel element performance is not limiting. Specific power (KW/ft.) is expected to increase in the order of 125 per cent, using the second

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quarter of FY 1959 as a base for comparison, and will vary by reactor type as shown in Figure VI.

Variable goal exposures for normal uranium I & E fuel elements are being adjusted to an average range of 700-750 MWD/T during power ascension. It is assumed fuel element performance will permit incremental increases by FY 1963 to 800 MWD/T. Enriched I & E fuel elements will approximate the same exposure levels.

Beginning in FY 1960, a program will be initiated to replace the present ribbed aluminum process tubes, as they are scheduled for replacement due to corrosion, with thin wall (30 mil vs 60 mil) ribless zircaloy-2 process tubes. This program is tentatively scheduled for completion by FY 1965. The larger diameter of the Zr-2 tubes will offer the potential advantages of higher coolant flow rates and increases in fuel diameter to improve reactivity. Fuel element designs will necessarily be altered to incorporate the self-support concept, which should minimize fuel misalignment and associated corrosion ruptures.

E. SUMMARY OF REACTOR TESTS

Objective

To evaluate the fabrication and irradiation of low hydrogen dingot fuel elements - PT-86-MT I & E.

To extend evaluation of the effect of optimum delay period (43 second) following Beta Heat-treatment, and prior to water quench on dimensional

Discussion

Six tubes of low hydrogen I & E dingot and 3 tubes of I & E ingot fuel elements were irradiated to an average exposure of 800 MWD/T. Post irradiation data show the average diameter growth of dingot fuel elements was 3.5 mils greater than that of ingot. Warp of the dingot averaged 9 mils greater than that of ingot. Some dingot fuel elements exhibited slight (3-5 mils) bumping.

Data from 6 tubes irradiated to an average exposure 585 MWD/T show no significant difference between warp and diameter changes of delay quenched rod heat-treated test material and rod heat treated controls.

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Objective

Discussion

stability of rod heat-treated fuel elements. PT 97-FP (16 tubes alternated test and control I & E Fuel elements, including 200 tubes for lot charging.)

An additional 8 tubes are continuing to a higher exposure goal.

Preliminary evaluation of direct cast I & E fuel element cores. PT-93-FP (12 tubes alternately charged cast and ingot I & E fuel elements.

Six tubes were discharged in March for Post irradiation measurements.

To evaluate further the effect of core blank heat treatment on dimensional stability of uranium fuel elements. PT 4-FP

Six measured tubes containing both core blank heat treated and rod heat treated cores were charged in March. Approximately 50,000 core blank heat treated fuel elements are currently undergoing irradiation to 800 MWD/T goal.

To evaluate lead-dip internally and externally (I&E) cooled tru-line fuel elements. PTs 73-MT, 79-MT, 80-MT, 83-MT, IP-141, IP-142, IP-168, IP-178 (lot charge)

During 1958 the average warp of I & E irradiated to normal goal exposure was approximately 10 mils. Diameter growth was nil. Approximately 1 per cent of I & E warp in excess of 40 mils with a maximum of 72 mils recorded.

To compare incidence of hot spots with bonding layer quality of enriched I & E fuel elements. PT: 1-FP

Thirty measured charges having three levels of bond quality are currently undergoing irradiation to 800 MWD/T goal. 240 tubes to be lot charged within two months for exposure to 800 MWD/T.

To evaluate corrosion resistance of of nickel-aluminum (1% Ni, .5% Fe) alloy components. PT's 68-MT, 84-MT, 91-FP, 2-FP and IP-124.

About 48% of all fuel elements in the reactors are clad in X-8001 alloy jackets. This includes 60% of I & E and 12% of solid. The rupture rate for X-8001 clad elements is a factor of 3 worse than C-64 clad I & E material in high flux reactors. However, ruptures were probably due to remedial process problems.

Evaluation of X-8001 alloy aluminum components fabricated from cast blanks. PT 6-FP.

Six measured columns of I & E enriched fuel elements clad in X-8001 alloy components from cast blanks and wrought blanks to be charged

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Objective

Discussion

To evaluate self-supported fuel elements in process tubes PT IP-84 (13 ribless tubes in B Reactor) PT 98-FP (bumper ribs for K Reactors)

in May. Subsequently, 240 columns of I & E enriched fuel elements clad in components fabricated from cast blanks will be irradiated for this evaluation.

Solid and I & E self-supported fuel elements irradiated to 400 MWD/T in B Reactor show no hot spots. 13 tubes are continuing irradiation in 100 MWD/T increments to 900 MWD/T.

Ten tubes of "K" size I & E fuel elements with bumper ribs (50 mils) with 10 tubes of controls were charged into standard "K" size tubes in March.

To evaluate the effect of Nickel plate on contamination of reactor effluent. PT IP-207.

Five charges, two electroplated, two chem plated, and one control to 800 MWD/T. One column with control metal was discharged below goal due to a reactor problem.

II. FUEL ELEMENT DEVELOPMENT

A. COEXTRUSION DEVELOPMENT

During March 1959, twenty-two rod billets were extruded at Nuclear Metals, Inc.

The extruded rods were clad with 0.020" thick zircaloy-2. Extruded sizes were 0.830", 0.820" and 0.622" O.D.

Variables introduced for evaluation in this test were:

1. Extrusion of billets fabricated at HAPO.
2. Dingot cores in the as-extruded and heat treated condition.
3. Uranium cores cast directly to size and heat treated.
4. Zircaloy-2 clad sleeves fabricated by centrifugal casting and cold extrusion.
5. High reduction ratios - 39:1.

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FIGURE II

* ONE LOT ONLY

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HW 5992182D

TABLE I
STATS PLUS SPECIFICATION PROJECTS
ALIGNED COLUMNS

LOTS	TYPE	DECEMBER		JANUARY		FEBRUARY		MARCH 1 1954	
		NO. OF PIECES	REF.	NO. OF PIECES	REF.	NO. OF PIECES	REF.	NO. OF PIECES	REF.
CA	Mixed Lots, Solid, N	2114	2.3	1498	1.1	1977	0.6	None Processed	
YA	FMPC Lots, Solid, E	None Processed		3907	1.3	1734	0.6	None Processed	
VA	Direct Lots, Solid, N	None Processed		None Processed		None Processed		None Processed	
KE	FMPC Lots, Solid, E	None Processed		880	1.2	None Processed		None Processed	
KG	FMPC Lots, 'C' I & E, E	19290	0.5	1695	0.3	6932	0.4	None Processed	
KH	FMPC Lots, 'O' I & E, E	None Processed		4329	0.4	5495	0.2	None Processed	
CJ	Mixed Lots, 'C' I & E, N	None Processed		5449	4.6	None Processed		None Processed	
LJ	FMPC Lots, 'C' I & E, N	None Processed		None Processed		None Processed		25331	1.1
CK	Mixed Lots, 'K' I & E, N	2019	1.9	None Processed		3281	3.8	2019	1.0
KK	FMPC Lots, 'K' I & E, N	5709	1.7	46016	0.2	5560	1.5	27727	1.3
KL	Mixed Lots, 'O' I & E, N	13024	1.0	8376	1.7	2578	2.2	None Processed	
SL	FMPC Lots, 'O' I & E, N	74012	1.2	43334	1.2	7970	0.6	14829	0.1

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TABLE II
SEAMS AND STRIATIONS
RECOVERED MATERIAL

LOT TYPE	DECEMBER		JANUARY		FEBRUARY		MARCH 1-15	
	NO.	%	NO.	%	NO.	%	NO.	%
	PROCESSED	REJ.	PROCESSED	REJ.	PROCESSED	REJ.	PROCESSED	REJ.
<u>Normal</u>								
"C" I & E	None Processed		12438	16.5	None Processed		2773	10.5
"K" I & E	30621	20.0	7127	14.2	3014	7.2	8177	12.1
"C" I & E	9315	13.4	1400	16.3	1403	10.4	1711	17.5
Solid	None Processed		5000	21.0	11100	17.3	None Processed	
							Avg.	15.2%
<u>Enriched</u>								
"C" I & E	None Processed		3840	5.8	1610	4.3	None Processed	
"K" I & E	None Processed		None Processed		None Processed		None Processed	
"C" I & E	1618	5.9	4370	5.9	6896	4.5	None Processed	
Solid	None Processed		3110	8.8	None Processed		None Processed	
							Avg.	5.8%

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TABLE III
POROSITY TYPE REJECTS

LOTS	DECEMBER			JANUARY			FEBRUARY			MARCH 1-15		
	BT	V	%	BT	V	%	BT	V	%	BT	V	%
	TOTAL			TOTAL			TOTAL			TOTAL		
CA	12.5	0.6	13.1	5.6	0.9	6.6	5.9	0.8	6.7	None Processed		
KA	None Processed			5.1	1.4	6.5	5.8	0.9	6.7	None Processed		
VA	None Processed			None Processed			None Processed			None Processed		
KE	None Processed			10.3	0.0	10.3	None Processed			None Processed		
KG	2.6	0.3	2.9	3.4	0.2	3.5	2.2	0.2	2.5	None Processed		
KH	None Processed			3.9	0.3	4.2	1.4	0.04	1.4	None Processed		
GJ	None Processed			16.1	0.3	16.4	None Processed			None Processed		
KJ	None Processed			None Processed			None Processed			1.0	0.3	1.3
CK	1.6	0.2	1.8	None Processed			9.9	0.7	10.6	14.1	0.2	14.3
KK	2.1	0.5	2.6	3.9	0.5	4.1	10.5	0.2	10.7	4.9	0.2	5.2
GL	2.8	1.0	3.7	3.9	0.4	4.3	2.3	0.5	2.9	None Processed		
KL	3.9	0.4	4.3	3.6	0.4	3.9	2.4	0.4	2.8	5.3	0.4	5.7

☐ Small quantities processed

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TABLE IV

COMPARISON OF CHLORIDE VERSUS CARBONATE HEAT TREATING

LOT TYPE	HEAT TREATMENT	QUANTITY	SONOTEST		INT. POROSITY		EXT. POROSITY		VOID	TOTAL POROSITY		
			#	%	#	%	#	%	#	%	#	%
KR	Chloride	12,560	1631	13.0	115	1.1	313	3.0	19	0.2	446	4.2
KR	Carbonate	7,348	209	2.8	25	0.3	118	1.6	8	0.1	145	2.0
KK	Chloride	24,891	3050	12.3	238	1.1	423	2.0	73	0.3	736	3.5
KK	Carbonate	18,720	509	3.6	125	0.7	499	2.6	64	0.4	648	3.6
KL	Chloride	78,746	7549	9.6	233	0.3	1570	2.2	275	0.4	2078	2.9
KL	Carbonate	100,648	2404	2.4	477	0.5	3347	3.4	398	0.4	4202	4.3

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TABLE V

REACTOR STABILITY OF ALPHA ANNEALED CORES

	LOW EXPOSURE (400)		HIGH EXPOSURE (720)	
	Regular	Alpha Annealed	Regular	Alpha Annealed
Warp	7.3	5.5	2.8	5.7
Diameter at End	-3.5	-3.3	4.4	-5.5
Diameter at Center	0.5	-0.1	0.1	-0.4
Tube Filling Capacity	5.4	3.3	7.2	2.8

UNITS: 0.001"

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TABLE VI

NONDESTRUCTIVE TESTING EQUIPMENT SCHEDULE

UAT	4/59 Prototype debugging and statistical sampling.	5/59 Prototype installation in production line, 313 Bldg. for statistical sampling and final debugging.	6/59 Prototype converted to final design in production use.	7/59 Final designs, standards and procedures available for FMPC.	9/59 Complete installation of 4 production test stations in 313 Bldg.
SGMT	Analyze accumulated data and define remaining variables to be investigated. Prepare reactor P.T. Begin development of frequency ratio readout and determine equipment improvements for production model	One prototype for statistical sampling.	Second tester for statistical sampling if needed.		Proceed with procurement of production testers when sufficient information justifies 100 percent use.
IDT	Conduct experiments with eddy current MGT to determine reliability in detecting extrusion defects.	If results are satisfactory, equipment could be available by May for direct core testing.			If results are satisfactory, equipment for 100 percent testing of all cores.
UST		Prototype installation in production line, 313 Bldg., to obtain frequency distribution and establish reject standards.	Prototype converted to final design in production use.	Final design, standards procedures available for FMPC.	Complete installation of 4 production test stations in 313 Bldg.
SGT	Conduct Feasibility Study.				
UVT	Continue cooperative study of UVT and SGT with FMPC				

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TABLE VII

TENTATIVE FUEL CORE PRODUCT SPECIFICATIONS

<u>UAT</u>		<u>MEYN INTERCEPT*</u>
GRAIN SIZE		
INGOT	MIN. AVG. TRANSMISSION, 12.0 Avg. to Peak Variation 30%	.300 MM Max.
DINGOT		
25 T/Mo.	MIN. AVG. TRANSMISSION, 4.1 Avg. to Peak Variation 100%	.400 MM Max.
60 T/Mo.	MIN. AVG. TRANSMISSION, 10.0 Avg. to Peak Variation 30%	.300 MM Max.

* These values are shown for comparative purposes only and are not yet considered absolute values in view of the limited data from which they were derived.

<u>CRYSTALLOGRAPHIC ORIENTATION</u>	<u>SORT</u>	<u>HAPC G.I.</u>
INGOT	>.7770	+ 1.0
DINGOT		
25 T/Mo.	>.7750	+ 2.0
60 T/Mo.	>.7770	+ 1.0

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FIGURE III

Self-Supported I & E Fuel Element. Eight equally spaced supports are spot welded to the aluminum jacket on each fuel element.

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FIGURE IV

Example of Spire Cracking with Moderate Amount of Al-Si Entering the Tube. 4X

FIGURE V

The Only Example of Partial Cracking Found to Date. The Sample was completely cracked through the spire wall at a different location as shown and Al-Si has filled the tube. 20X

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Figure VI

Based on preliminary evaluation work completed to date, the following results were obtained:

1. Billets Fabricated at HAPO

Ten of the twenty-two billets were assembled at HAPO, evacuated, shipped to NMI, and extruded. The bond tests on this material showed good bond quality.

Billet assembly consisted of:

- a. Etching the uranium core in HNO_3 .
- b. Scrubbing the machined zircaloy-2 clad in alkaline solutions.
- c. Etching and outgassing the copper cans.
- d. Evacuating with a helium leak detector.
- e. Sealing off using a bolt cutter to pinch the tube and cutting off the pinch-off with a heliarc torch.

During assembly of the billet particular care was exercised to prevent contamination of the components. The billet was flushed with argon gas during the final weld closure of the copper can and evacuated immediately after completion of the weld. Final vacuum was 1×10^{-5} mm Hg at seal off. Average pump down time was about 30 minutes.

2. Dingot Cores

Dingot material was prepared by gamma extrusion to seven inch rod and alpha extrusion from seven inch to 3.9 inch. Two billets were machined directly to final size and extruded in this condition. Both exhibited extreme surface and interface roughness. The clad was heavily grooved throughout the length of the extrusion, and some penetration of uranium at the interface was present. Reduction ratio of this material was 22:1.

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Four billets were rough machined to size, triple beta heat treated and machined to final size. About 0.030" per surface was removed by final machining. Two billets were extruded at 22:1 reduction. The clad surface was smooth, but there was some roughness evident at the interface. The material is considered to be of good quality, but the interface can probably be improved. Two billets were extruded at 39:1 reduction ratio. This material showed the same characteristics as at the lower reduction with slight improvement of the interface.

3. Cast Cores

All cast cores were machined to rough size, triple beta heat treated and machined to final size. All material exhibited good surface and interface characteristics. The core-clad interface of the 39:1 reduction ratio was exceptionally smooth and uniform.

4. Cast and Extruded Zircaloy Clad

This material was fabricated by centrifugally casting hollow billets and cold (7-800 F) extruding them at a 4:1 reduction ratio. This material showed a uniform rough outer surface and a very smooth interface. The outer surface showed the same structure as that encountered with beta treated zircaloy, but the roughness was reduced by a factor of 2-3. The surface will be acceptable after bright etch and swaging.

5. High Reduction Ratios

Since no material of this geometry had been produced at high reduction ratios, a 39:1 reduction was used to evaluate the effect of increased reduction. This material showed improved clad-core interface characteristics. All other

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factors looked to be equal to or better than low reduction material of the same metal history. A larger as-extruded grain size is apparent. The actual effect of the reduction ratio, if any, will not become apparent until detailed evaluation is complete.

6. General Evaluation

a. Dimensions

The as-extruded dimensions corresponded closely with calculated values. Variation along the length of the extrusion was not excessive, with total variation along 85% of the extruded length falling within a $0.005'' (\pm 0.0025'')$ range for each rod size.

b. Yield

Based on initial evaluation the yield of acceptable rod uranium weight to billet uranium weight will be about 80%. This is with a billet design giving a billet length to diameter ratio of 2-1/2 to 1.

B. FABRICATION AND TESTING OF NEW FUEL ELEMENTS

1. Bond Strength and Defect Tests of Coextruded Fuels

The bond strength of coextruded Zircaloy-2 clad uranium fuel rod as effected by various heat treatments has been determined. Heat treatments which produce a thickened bond zone reduce the shear strength of the bond.

A resume of the data follows:

<u>Sample Condition*</u>	<u>Bond Shear Strength - PSI</u>	<u>Remarks</u>
As extruded	Avg. of 3 = 69,250	Failure by shear in the cladding or the U adjacent to the bond
	Avg. of 3 = 55,000	Failure by tension in the cladding away from the bond

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<u>Sample Condition*</u>	<u>Bond Shear Strength - PSI</u>	
Salt bath beta treated, 720-730 C 10 min., water quench	Avg. of 2 = 71,000	Failure by shear in cladding adjacent to the bond
Salt bath beta treated, 720-730 C 10 min., air cool	Avg. of 2 = 61,050	Failure by shear in cladding with some tendency to fail in bond
Salt bath beta treated, 720-730 C 10 min., quench to 590-600 C, hold 10 min., air cool	Avg. of 3 = 57,775	Failure partially by shear in cladding with some tendency to fail in bond
Vacuum furnace beta treated, 720-730 C, nominal 10 min., furnace cool	Avg. of 2 - 47,750	Failure predominately in bond
Vacuum furnace diffusion treated, 62 hours, 840 C, furnace cooled	27,800	Failure in bond
Vacuum furnace diffusion treated, 62 hours, 840 C, furnace cooled, 2 hrs., 840 C, oil quenched	12,870	Failure in bond

*Sample 0.630" dia., 0.030" clad; Uranium bored out of one end and cladding turned from other end leaving 0.040"-0.050" bonded zone - sample pulled in tension applying shear to bond zone.

Coextruded fuel shapes are being defect-tested to determine their failure behavior and fuel corrosion rates under various test conditions and in various heat treated conditions. These studies are being made in an autoclave, an ex-reactor flow loop, and in a resistance heated fuel rod flow loop facility. Autoclave tests are being made in an induction heated autoclave so that the effect of slow heating and cooling upon the defected test specimen can be eliminated. Autoclave test specimens have been uniformly defected with a 0.025" diameter hole through the cladding and just into the uranium. The results of these autoclave tests indicate that a high bond strength may not necessarily be the

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most desirable from the viewpoint of failure rate and fuel rod distortion accompanying failure. "As extruded" specimens with uranium fuel cores have been the most resistant to failure in the early stages of failure, that is, when the initial defect is growing but the specimen has not yet ruptured or lost significant uranium. However, after failure has once been initiated by rupturing of the cladding, weight loss accompanied by severe clad distortion is usual. Beta treated and water quenched specimens fail the most rapidly during the initial stage of failure and also distort badly during failure. There have been some anomalous behavior observed in this condition, however, and this may indicate that the mechanism of failure is more pertinent than the bond strength of the specimen.

Ex-reactor flow loop tests are being made on specimens in various heat treated conditions and with various types of defects. The results of these tests to date indicate that the highest bond strength is not the most desirable, but rather strengths produced by slow cooling treatments seem to perform best.

Tests are continuing, and include, in addition to coextruded uranium rod, coextruded uranium tube sections and coextruded U-2 w/o Zr alloy rod.

2. Cluster Fuel Elements

The cluster fuel geometry is one of two geometries being considered for Hanford's New Production Reactor. As part of a development program, fourteen 7 rod cluster fuel elements are presently being irradiated in both Hanford and ETR high temperature loop facilities. These elements are

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fabricated from rods of uranium coextruded and clad with 20 or 30 mils of Zircaloy-2. In the KER Loop 1 facility are seven such elements enriched to 1.6% U235, three having 30 mil cladding and four with 20 mil cladding. Goal exposure for these elements is 4500 MWD/T. Four additional 1.6% enriched elements with 30 mil cladding are being irradiated in the KER loop 2 facility to a goal exposure of 3500 MWD/T. Operating temperature of the loop coolant is 230 C bulk outlet in both cases. Three natural uranium 7 rod cluster fuel elements are being irradiated in the high temperature loop facility in the ETR to a goal exposure of 1000 MWD/T. Bulk outlet coolant temperature is 280 C.

Four 7 rod cluster fuel elements clad unbonded in stainless steel were discharged in February, 1959 after 3500 MWD/T exposure. These elements were instrumented to gain specific power and core temperature data. From such data the heat transfer bond coefficient between the uranium and stainless steel cladding was calculated as a function of burnup. At the beginning of the test the average core temperature was observed to be 522 C and the average specific power was calculated to be 119 kw/ft. The bond coefficient was calculated as 1300 BTU/hr-ft²-°F. Prior to discharge the corresponding values were 280 C core temperature, the specific power was 80 kw/ft, and the bond coefficient had risen to 3000 BTU/hr-ft²-°F. Post irradiation examination is now being started.

Seven rod cluster elements have been irradiated to 1250 MWD/T in KER Loop 1. The maximum bulk water temperature was 270 C. All fuel rods were

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coextruded Zircaloy-2 clad natural uranium. Average specific power in the fuel was 12 watts/gm. One element was designed with a different spacing between the six external rods. The inter-rod gap varied from .025 to .055 inch.

After irradiation only the most closely spaced rod showed significant diameter increase. All rods increased slightly in length. None of the rods in the one element examined suffered warp.

DIMENSIONAL CHANGES IN SEVEN ROD CLUSTER ELEMENTS EXPOSED TO
1250 MWD/T IN 270 C WATER

Rod No.	Length		Center Diameter		Max. Core Temp. (Calculated)
	Pre	Post	Pre	Post	
9	13.218	13.245	.6315	.631-.632	440-460 C
10	13.213	13.228	.633-.634	.634-.635	450-470
12	13.230	13.234	.631-.632	.630-.632	450-470
14	13.232	13.238	.632	.639-.640	500-520

The preceding table shows the changes in four fuel rods. Rod 14 was a closely spaced rod with core temperature about 50 C higher than adjacent rods. All rods have .030 inch Zircaloy-2 clad.

Four four-rod cluster fuel elements fabricated from 1.6 per cent enriched coextruded uranium-Zircaloy-2 rod were discharged from the KER Loop 2 facility in November, 1959, after an exposure of 0.14 a/o burnup. Post irradiation examination showed that macro-cracks existed in all sixteen rods. Only in one of the sixteen rods did cracking occur in the bond region. This cracking was random throughout the microstructure, existing both across and along grain boundaries. Measurements after irradiation revealed

changes in outside diameter up to 0.012 inches and in length up to 0.182 inches.

This corresponds to a 4 per cent volume increase or a R value of 29.

Further examination will continue to determine the relative amount of swelling and cracking in the ends of the rods as compared to the central sections already examined.

3. I & E Cold Closure

Attempts are being made to cold close I & E fuel elements by cold pressure welding the internal and external aluminum jacket material together in an annular ring joint at the end of the element core. The cladding is sized onto the O.D. and the I.D. of the core so that rings of cladding material extend past one end of the core. The element is placed in a container and a mandrel inserted through the I.D. The concentric rings of material are upset laterally and pressed together to form the ring joint by a punch designed with an annulus above the joint so that the metal in the joint may extrude out through the punch. The sidewise extrusion of the material at the ring joint is necessary to break up the oxide surface in the joint region which otherwise prevents the metal from pressure welding. Cold pressure welds are not possible unless the aluminum is extruded down into a small annular groove and extruded up through the punch simultaneously so a small annular groove is required in the end of the core below the ring joint.

Work has consisted of: (1) obtaining a cold pressure weld in ring joints which simulate the joint at the end of the element core; and (2) sizing the cladding onto the internal and external surfaces of the core. The closure on an

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element will be attempted when sizing difficulties are overcome.

4. Coextrusion of Zircaloy Clad Uranium Rod

The extrusion program has been continued to study variables in billet preparation, billet component fabrication, and extrusion conditions as applied to the coextrusion of Zircaloy-2 and uranium. Additional tests were made employing the 280 ton press in the Plutonium Fabrication Laboratory. The billet designs and component fabrication were adjusted to learn the influence of copper clad thickness and uranium structure on the uranium-Zircaloy-2 interface. The billets were sealed, both Zircaloy-2 and copper, by the electron beam welding process. Extrusions were made at a reduction ratio of 10:1 with the billet preheat at 650 C, die preheat 550 C, and container heated to 575 C. The following table shows the billet conditions that were extruded.

<u>Billet No.</u>	<u>Copper Clad Thickness- inch</u>	<u>Uranium Billet</u>	<u>Preheat Time Hrs. @ 650C</u>	<u>Extrusion force-tons</u>
3	0.032	Beta heat treated dingot	1-1/2	175
1	0.063	Alpha rolled dingot	3	275
2	0.063	Beta heat treated dingot	3-2/3	260

The rods as extruded were 0.637-0.639 inches in diameter. After copper removal all three rods were approximately 0.615 inches in diameter. Transverse sections of the lead and trail ends of each extrusion are shown in Figure VII. The improved Zircaloy-2 uranium interface for the as-rolled core material is apparent. Reducing the copper clad thickness on the billet also results in a smoother interface but the effect is not as pronounced as that produced by the uranium structure. All the rods appear well bonded. There are wide

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variations in the as-extruded uranium grain size both from near the clad to the center of the rods and from end to end of the rods. The material nearest the clad was consistently larger in grain size but the variation along the length was not predictable.

Sections of billets No. 2 and No. 3 were cold swaged after extrusion to intermediate steps ranging from 0.600 to 0.580 inches diameter. Examination is being made of these rods as swaged and after beta heat treatment to determine the effect on the interface, clad thickness, and uranium structure.

A series of billets were prepared for extrusion at Nuclear Metals, Inc. during the first week of March. Seven billets were prepared, including three machined uranium assemblies and four assemblies made by casting the uranium directly into heated Zircaloy-2 billet shells. The machined cores were made from 4-inch diameter dingot uranium that had been alpha extruded from a 7-inch diameter gamma extrusion. Two of these cores were vacuum outgassed 12 hours at 600 C and the third was assembled without further treatment. Two of the direct cast billets were given a single beta heat treatment of 20 minutes at 730 C in Nu-Sal salt and water quenched. Two direct cast billets were extruded as cast. All the billets were clad with 0.063 inch thick copper jackets, 3.000 inches in outer diameter. Extrusions were made at 21:1 extrusion ratio with a billet preheat of 1155 F. The resulting rods have been examined for Zircaloy-2 surface, Zircaloy-2-uranium interface and limited examination has been made of the bond zone. All the machined dingot uranium cores produced an extremely irregular interface due to large gamma domains not broken up by the 3:1 alpha extrusion. The macrostructure of the rod from which these cores were machined is shown in Figure VIII. The as-cast billets also produced a rough, irregular Zircaloy-2

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uranium interface, probably due to the large as-cast uranium grain size. In addition, the outer Zircaloy-2 surface was rough with sharp longitudinal ridges. This effect is attributed to the Zircaloy-2 structure produced during casting the uranium into the billet cup. The Zircaloy was heated into the beta phase to about 1200 C maximum and upon cooling had the typical large grained beta treated structure shown also in Figure VIII. The cast and beta heat treated billets produced a much smoother Zircaloy-2 uranium interface but also had the rough outer Zircaloy surface. Cross sections of three rods illustrating these effects are shown in Figure IX. The as-extruded bond zone of the cast and cast plus beta heat treated material is approximately the same as a normal coextrusion followed by beta heat treatment. The thicker diffusion zone is not entirely uniform at the sections examined to date as shown in Figure IX.

5. Coextruded Fuel Rod Closures

A new closure process has been developed to meet the requirements for high quality end closures on coextruded Zircaloy-2 clad fuel rods suitable for NPR cluster fuel element loading. This consists of forming the ends of the fuel rod by a semi-extrusion operation with a special die in a press. All rod components are heated to 620 C and a special punch preferentially extrudes the uranium core through a 3/16 inch diameter hole while thickening the Zircaloy-2 clad sufficiently to cause the Zircaloy to close over the end of the rod to within 1/8 inch of the center. Following the nosing operation the excess metal is faced off and a ring type resistance projection weld is made on the thickened nose. The projection welding operation is fast, simple and yields consistently high

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quality welds as shown in Figure X. After projection welding the element is tested for jacket and weld integrity by a short time 400 C steam autoclave test. Following this test an electron beam fusion weld is placed between the cap and the thickened shoulder of the fuel rod.

A double closure of extremely high quality is produced with the closure free of uranium contamination and satisfactory for long time in-reactor exposure.

6. The Leak-Vulnerability Test

A test that is useful in evaluating the suitability of new designs of fuel elements having uranium cores is the Leak-Vulnerability Test, a modification of the Water-Autoclave Test, sometimes called the "Undercutting" Test.

This test gives a general idea of the manner in which fuel elements of the type being tested would behave during reactor irradiation if a small leak were to develop in the primary jacket due to mechanical injury or corrosion pitting. It shows something about the effectiveness of the secondary barrier layer in blocking or delaying the action of invading coolant water, and indicates whether or not the core material is relatively resistant to attack by water.

The test consists of exposing fuel elements with intentional jacket defects to the action of water at temperatures around 200 C, and making frequent examinations to determine the progress of attack by the water on the core material. The jacket defects are carefully introduced to permit entry of the

water to the interface between the jacket and barrier layer at identified points, and completely through the barrier layer at other points, so that an idea of the effectiveness of the barrier layer may be gained. Metallographic examination of the test specimen aids in verification of the mechanism by which the attack was propagated.

While this test is useful in estimating the general quality of a product with respect to its vulnerability to certain unfavorable reactor conditions, its very nature makes it highly subjective and hence, not well adapted to use as an acceptance test for general production inspection. However, it can be used to advantage in categorizing lots of fuel elements to be charged into the reactor, those lots which are shown to be especially vulnerable in this test may be exposed less rigorously in the reactor, to the end that probability of in-reactor failure may be reduced.

7. The "Ping" Test

Fuel elements canned with AlSi (and perhaps other) bonding media are subject to bond fracture under relatively light stress, unless the conditions of processing are controlled within fairly narrow limits. A non-destructive test to indicate the frangibility of bond has been developed to aid in the evaluation of new fuel element processes and designs.

This test is based on the principle that, when heated, differential thermal expansion between the core material and the jacket material (usually aluminum or aluminum alloy) will cause the jacket to break loose from the core if the adhesive strength of bond is less than the yield strength of the jacket material.

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Such breaking takes place ordinarily in a piecemeal fashion, a small increment in strain being accompanied by a small increase in fractured area. The relief of stresses when brittle fracture occurs creates a snapping sound which may be heard sometimes without aid; with suitable sound pickup and amplification, it becomes easily audible.

The test procedure consists of heating the fuel element rapidly by various means (in which no sound is produced by the act of heating itself), contacting the fuel element with a pickup, and amplifying the sound of the fracturing bond so that it may be heard distinctly as a series of "pings" in a pair of earphones. A necessary precaution is the shielding of the system from extraneous sounds which may mask or overpower the desired signal. This test is currently being evaluated for reliability, ease and speed of operation.

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Lead and Trail End - Beta Heat
Treated Dingot Core - 0.032-inch

Lead and Trail End - Beta Heat
Treated Dingot - 0.063-inch

Lead and Trail End - As Rolled
Dingot Core - 0.063-inch Copper.

Upper and Lower Right

Upper and Lower Middle

Upper and Lower Left

FIGURE VII. TRANSVERSE SECTIONS OF CO-EXTRUDED RODS - 3X

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Macrostructure of 4-inch Diaget Uranium
Alpha Extruded from 7-inch Gamma Extrusion IX.

Structure of Zircaloy-2 Billet Jacket
After Uranium Casting - 100 X.

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FIGURE VIII. STRUCTURES OF EXTRUSION BILLET COMPONENTS

Transverse Sections of Co-extruded Rods - 3X.
Upper - Machined Dilect Uranium Core; Lower Left - Direct Cast
and Beta Heat Treated; Lower Right - Direct Cast.

Zr-2

Two Areas of Direct Cast and Extruded Rod - 250X.

FIGURE IX. ZIRCALOY-2 CLAD URANIUM EXTRUSIONS

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FIGURE X. CROSS SECTION OF CO-EXTRUDED U-2R-2 FUEL ROD
WITH CAP ATTACHED BY PROJECTION WELD AND ELECTRON BEAM
VACUUM WELD - MAGNIFICATION 9X

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