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August 29, 1961

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

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REPORT TO THE WORKING COMMITTEE

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REPORT TO THE WORKING COMMITTEE

From

General Electric Company

Hanford Atomic Products Operation

Richland, Washington

J. T. Stringer

and

J. E. Minor

August 29, 1961

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TABLE I
METAL QUALITY REBATES

	<u>April</u>		<u>May</u>		<u>June</u>		<u>July</u>	
	<u>Virgin*</u>	<u>Recovered</u>	<u>Virgin</u>	<u>Recovered</u>	<u>Virgin</u>	<u>Recovered</u>	<u>Virgin</u>	<u>Recovered</u>
<u>NORMAL LINGOT U</u>								
No. of pieces processed	155,533	14,690**	133,025	4,875**	131,542	42,833**	54,024	5,779**
Percent rejected	0.07	1.0	0.05	1.8	0.2	1.7	0.1	1.3
<u>NORMAL LINGOT U</u>								
No. of pieces processed	23,966		19,111		39,147		36,032	
Percent rejected	2.08		0.05		0.05		0.08	
<u>REFINED LINGOT U</u>								
No. of pieces processed	52,256	2,665	71,218***	2,186	16,034	8,050	17,219	8,420
Percent rejected	1.1	11.6	0.8	7.0	0.2	8.0	0.4	4.7

* Virgin reject rate for April does not include UT-2 surface test rejects.

** Recovered lingot cores are included in recovered normal ingot cores.

*** Includes some 400 Series metal. Includes some 400 Series metal.

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Figure 1. Average Grain Size of I & E Uranium Cores Measured at HAPC
vs. Shipping Date from Feed-Sites.

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TABLE II

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AVERAGE TOTAL HYDROGEN VALUES OF AS-RECEIVED CONES
April '61 - July '61

<u>Core Model</u>	<u>Lot Type</u>	<u>Type Metal</u>	<u>Average Total H₂, ppm</u>	<u>Range, ppm</u>	<u>Total No. Samples</u>
CIIN	KJ	Normal NLC Ingot	1.4	0.97-2.1	1
CIIN	KJ	" " "	1.3	0.90-3.5	1
CIIN	KJ	" " "	1.4	0.92-1	1
CIIN	KJ	" " "	1.5	1.0-1.8	1
CIIN	KD	" " "	1.6	1.5-1.7	1
CIIN	KI	" " "	1.3	1.1-1.5	1
CIIN	P2,V2,H	Regular Dingo	1.0	1.1-2.0	1
CIIN	Q2	Vacuum Outgassed Dingo	0.80	0.40-1.5	1
CIIN	UV	" " "	0.90	--	1
CIIN	Q2	" " "	0.75	0.61-1.3	1
CIIN	WZ	Vacuum Outgassed Extruded Dingo	1.1	0.95-1.6	5
CIIE	KY	Enriched Ingot 400 Series	1.7	1.2-1.9	1
CIIE	KY	Enriched Ingot 500 Series	1.7	1.0-2.3	20
KIVE	KK	Enriched Ingot 400 Series	1.9	1.5-2.2	2
KIVE	KK	Enriched Ingot 500 Series	1.9	1.1-2.3	20
CIIE	KC	Enriched Ingot 400 Series	1.6	1.3-2.1	3
CIIE	KC	Enriched Ingot 500 Series	1.7	1.2-1.9	3

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TABLE III

POROSITY DATA AS TOTAL EXTERNAL BOND COUNT
FOR APRIL THROUGH JULY 1961

Total External Bond Count						
Lot Type	Core Model	No. Pieces Measured	>100 Counts	> 20 Counts	> 5 Counts	
<u>Normal Ingot</u>						
<u>Virgin</u>	KF	KIHH	5717	4.5	1.5	0.5
	KZ	OIHH	1431	2.7	7.2	25.1
	KJ	CIIN	678	0.9	2.3	27.6
		<u>5826</u>	<u>8.1</u>	<u>11.0</u>	<u>53.2</u>	
<u>Recovered</u>						
	CF	KIHD	402	1.0	10.0	12.0
	CE*	OIHE	284	1.1	1.1	1.1
		<u>686</u>	<u>2.1</u>	<u>11.1</u>	<u>13.1</u>	
* Includes recovered ingot cores						
<u>Normal Dingo</u>						
<u>Virgin</u>	PZ,VZ,HZ	OIHH	486	2.0	3.1	4.1
	UZ*	OIHH	341	5.0	16.7	49.0
	UZ**	OIHH	108	0	0.0	13.0
	LZ***	OIIN	95	2.1	1.1	13.7
		<u>1030</u>	<u>9.1</u>	<u>21.9</u>	<u>80.7</u>	
* Vacuum outgassed before heat treatment						
** Vacuum outgassed after heat treatment, Category 1.						
*** Large grain lot containing UZ,PE,VZ & HZ material.						
<u>Enriched Ingot</u>						
<u>Virgin</u>	KK	KIVE	692	0.0	0.0	31.0
	KY	CIHE	1946	0.1	2.1	21.0
	KC	CIHE	196	3.1	11.0	11.8
		<u>2834</u>	<u>3.2</u>	<u>13.1</u>	<u>63.8</u>	
<u>Recovered</u>						
	KK,KC & KE		210	0.0	7.0	2.0

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and (4) Increased use of pressurized quenching equipment with complete conversion in July. The virgin cores having the lowest total bond count were the vacuum-outgassed-after-beta-heat-treatment dingot cores. There were no appreciable changes in bond porosity levels in recovered core models since the last reporting period.

There were no significant changes in normal ingot and dingot core rate of change from the last reporting period. However, the reactivity values for oxidized ingot cores has increased, as shown in Fig. 2, when the conversion was made to 500 Series metal. The slight increase in CIIIE and KIVE reactivity for the 500 Series metal is believed due to decreases in impurities, but the large increase in CIIIE reactivity is presently unknown. The tolerance of 500 Series ingots has been relaxed on a wider basis (6 months) from $0.947 \pm .006\%$ to $0.947 \pm .010\%$, but all of the CIIIE cores were well within the previous $\pm .006\%$ tolerance.

The adoption and canning of 0.053 inch longer cores in all major production models was accomplished during the reporting period. The increase in core length is made without changing the nominal over-all fuel element length or reducing the average aluminum cap and base thickness.

Dingot Uranium

As of July 31, 1961, 306 tons of additive dingot uranium were undergoing irradiation and 664 tons had been discharged. A summary of the dingot failures that occurred during the reporting period is as follows:

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Figure 2. Enriched Ingot Core Reactivity Measurements vs. Enrichment Data from Feed-Sites.

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Dingot Failure no.	Failure Date	Lot No.	Power		Exposure MWD/T	Rupture Type
			KW/tube	KW/ft.		
6	4-28-61	OL-010D	631	27	760	Internal
7	5-1	LZ-500A	1268	54	491	Side Hot Spot
8	5-2	LZ-500A	1167	50	450	Unclassified
9	6-9	UZ-014A	977	44	305	Transverse Split
10	6-17	VZ-006B	1171	50	292	End (Core)
11	6-24	UZ-015H	1237	56	208	End (Core)
12	7-1	VZ-015E	1172	53	210	Transverse Split

Sixteen failures have occurred to date in the ingot control material (3 side hot spots, 3 side others, and 1 unclassified). In order to prove that dingot fuel performance is two-fold worse than ingot, under the present test design, approximately ten additional dingot failures must be experienced without incurring any failures in the ingot control material. Conversely, about six additional ingot failures in the control material without incurring any dingot ruptures are required to demonstrate the performance of dingot is at least equivalent to ingot material.

The two transverse split dingot failures are the first split failures to occur in natural I & E fuel elements. Different dingot numbers were observed on the two transverse split failures and there was no evidence of ~~gross~~ cracks on the core surface. Metallographic examination of the first split failure in Radiometallurgy has not revealed the exact failure mechanism. However, both transverse and longitudinal cracks originating at the core I.D. were observed as shown in Fig. 3. This suggests that the pressure of uranium oxide buildup noted in the core I.D. contributed to at least part of the cracks. It is possible that small cracks in the center of the solid core blank, resulting from the volume change during the beta-alpha transformation, were not completely removed by core drilling, and the remaining cracks produced serious stress risers that resulted in the failure or the cracks were extended by oxide buildup resulting from water entry from elsewhere in the assembly. Examination of the second dingot failure is continuing in an attempt to more precisely define the failure mechanism.

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Figure 3. First transverse split failure in an additive during a GRIEN fuel element. About a 1/2 inch section is missing from the center of the element. Note transverse and longitudinal cracks radiating from the I.D. Target No.: UZ-57512. Exposure: 305 MWD/T. Power: 44 KW/ft.

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Two recovered dingot cores from the same dingot bar were found at pickle inspection that had gross cracks on a portion of the surface, as shown in Fig. 4. One core was destructively examined at HAP0 and the other by MCW. The cracks were found to be up to 1/4 inch deep. Uranium hydride lined the cracks and extended into the metal adjacent to the cracks in the form of hydride whiskers. These cores had been vacuum outgassed before bath heat treatment. It is not known when and how the hydrides were formed or whether the hydrides caused the cracks or were formed in the cracks after the cracks had formed from some other mechanism. The dingot lots which sustained the two transverse split failures were known to contain cores from the dingot that produced the cracked cores but the dingot numbers of the failures were different than the numbers of the cracked cores. Plans for increasing the amounts of dingot inspection were made at HAP0 from a nominal 100 ± 20 tons per month to unlimited use and were deferred until the current failure and cracking problem has been fully investigated. However, the conversion from QIIN to KIVN model is being made to eliminate the confusion in having dingot cores included in the Aluminum Vendor and the Buyer Programs.

A large scale test of about 40,000 additive dingot cores was run by MCW to vacuum outgassing hollow core blanks and finished cores after bath heat treatment compared with the standard practice of vacuum outgassing solid core blanks before heat treatment. Vacuum outgassing in hollow form after bath heat treatment appears advantageous since 1, it requires a shorter outgassing time, 2, the hydrogen absorbed during the salt bath heat treatment is removed and 3, the outgassing temperature is in the stress anneal range and should also relieve the residual quench stresses. The test consisted of the following three treatments:

1. Category 1, consisting of about 10,000 cores outgassed in a low total hydrogen content (less than 1.0 ppm) in hollow core blank form with a finished I.D.

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Figure 4. Surface appearance of cracks observed on two recovered additive dingot cores from Dingot No. UZ-5673.

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2. Category 2, consisting of about 10,000 cores outgassed to a low total hydrogen content (less than 1.0 ppm) in final core form.
3. Category 3, consisting of about 20,000 cores outgassed to a low surface hydrogen content (less than 0.4 ppm in the outer 40 mils) in final core form.

All of the Category 1 cores and a portion of the Category 2 cores have been tested to date, with completion of canning expected in Sept. 1961. From canning results to date, the cores vacuum outgassed after heat treatment appear to have less braze porosity than dingot cores outgassed before heat treatment. As shown in Table III, the total bond count indicates that Category 1 cores have less braze porosity than all other types of dingot cores evaluated. A comparison of the bond tests and the pin hole reject rates to date are as follows:

<u>Core Type</u>	<u>Approx. No. Cores Bond Tested</u>	<u>External Bond Reject Rate</u>	<u>Internal Bond Reject Rate</u>	<u>Pin Hole Reject Rate</u>
Category 1	3800	0.5%	0.5%	0.2%
Category 2	3620	0.6	0.2	0.02
Category 3	4800	0.4	0.06	0.01
Control*	11030	0.5	0.8	0.1

* Vacuum outgassed before heat treatment

Uranium Core Testing

The number of OIIN and KIVN virgin cores being segregated into large grain lots (less than 0.3 minimum voltage) by the UT-2 testers is steadily decreasing as shown in TABLE IV. This decrease is due to improved UT-2 tester performance and a decreasing grain size (as indicated in Fig. 1) which is attributed to an increase in iron and silicon content in ingot metal and better control of quench conditions during heat treatment. Of interest, is the high large grain segregation rate (22.8%) of the rebored OIIN cores that were shipped initially to HAPC in Nov. and Dec. 1960, when the feed-sites were experiencing a large grain

TABLE IV

UT-2 GRAIN SIZE TEST DATA

	<u>Large Grain Segregation (less than 0.3 min voltage)</u>			
	<u>APRIL</u>	<u>MAY</u>	<u>JUNE</u>	<u>JULY</u>
<u>Ingot - OIIN & KIVN</u> Percent Segregated	10.4	4.8	3.4	0
<u>Dingot - OIIN</u> Percent Segregated	6.3	8.5	4.0	0.4
<u>Mixed Ingot & Dingot - OIIN*</u> Percent Segregated	--	--	22.8	--

	<u>Large Grain Rejects ** from Segregated Lots</u>			
	<u>APRIL</u>	<u>MAY</u>	<u>JUNE</u>	<u>JULY</u>
<u>Ingot - OIIN & KIVN</u> No. cores tested	15,292	3870	6022	0
Percent rejected	0.02	0	0	-
<u>Dingot - OIIN</u> No. cores tested	2,339	0	3,300	0
Percent rejected	0	-	0.09	-
<u>Mixed Ingot & Dingot - OIIN*</u> No. cores tested	-	-	2,396	1,484
Percent rejected			0.42	0

	<u>UT-2 Grain Size Rejects** on all core material other than OIIN, KIVN & CVIN</u>			
	<u>APRIL</u>	<u>MAY</u>	<u>JUNE</u>	<u>JULY</u>
Normal Ingot	0	0	0	0
Enriched Ingot	0	0.07	0	0.28

* Mixed core lots were obtained from rebaring OIIN cores left over from the OIIN phase out in Dec. 1960.

** Includes fine grain, difference and large grain (less than 0.1 minimum voltage) rejects

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problem in normal uranium cores (as shown in Fig. 1). As of July 31, 1961, 97 tons of large grain segregated cores were being irradiated and 91 tons had been discharged. Two failures have occurred to date in a large grain dingot lot and were classified as a side hot spot and an unclassified. Neither failure could be attributed directly to metal quality.

To aid in controlling the grain size of cores, a minimum specification of iron plus silicon content of 100 ppm is being established for all uranium cores. Out of 312 KIVN cores rejected at 0.3 minimum voltage by the UT-2 testers in early Aug., 63% were from ingots having less than 100 ppm of iron plus silicon. Grain size of A-1 is common in 500 Series enriched cores that are fabricated from ingots that did not have iron and silicon added to the reheat charge. With this minimum specification, it is expected that all uranium cores with a grain size less than A-3 can be rejected without experiencing a prohibitively high reject rate.

During the reporting period, two incidents of untransformed cores occurred:

- a. Thirteen untransformed vacuum-outgassed-before-heat-treatment dingot cores supposedly heat treated by NLO were rejected by the UT-2 testers. These cores were in an OIIN lot that had been re-bored to OIIN size and were supposedly heat treated last Oct.
- b. Eight untransformed vacuum-outgassed-before-heat-treatment dingot cores supposedly heat treated by MCW in June were also rejected by the UT-2 testers.

The O.D. surface test circuit on the UT-2 testers was removed for further development work in mid-April. The test circuits had been installed in Jan. before the development work was complete, since three incidents of gross quench cracks (mud-flat cracks) in ingot uranium had occurred as a result of high iron contamination in the ingots. A test circuit to detect "mud-flat" cracks is currently being evaluated on a production UT-2 tester and the installation on all UT-2 testers is expected by the end of Aug. 1961.

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CVIN Cores for Overbored Process Tubes

The CVIN extruded dingot cores (379 pieces) for replacement columns in the 20 overbored process tubes were received, bare core tested, and canned during July. All the cores were vacuum outgassed after beta heat treatment in hollow blank form. The average grain size of this shipment is finer than the previous extruded dingot cores received in Nov. 1961. A lower reject rate for internal and external surface defects. The comparison of the bare core inspection reject rates for the two extruded dingot shipments are as follows:

<u>UT-2 reject rates</u>	<u>Shipment Date</u>	
	<u>July 1962</u>	<u>Nov. 1961</u>
Large grain	0.1	1.1
Difference (grain variation)	0.7	0.2
Fine Grain	0.5	0.7
<u>Surface test reject rates</u>		
Internal	2.7	0.6
External	7.0	--
<u>Visual pickle inspection</u>		
Reject Rate	0.9	0.2

An additional 18 tons of extruded dingot CVIN cores for charging into 32 new overbore tubes now being installed in J Reactor is expected to be shipped by MCW in August 1962.

Alpha Extruded Cores

The evaluation of alpha extruded cores, as an alternative to rolling and purposed cheaper method, will be carried under two separate test programs:

1. A well designed statistical test utilizing cores from eight ingots and four dingots, with bars A and B being used in each dingot. The irradiation of 32 tubes in a serial (16 fuel elements from each ingot and dingot bar) in a known pattern will give the following

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comparisons: rolled vs. extruded, ingot vs. dingot, bar A vs. bar B, difference among dingots, difference among ingots, whether the effect of extrusion is the same for ingot and dingot, and some reactor differences based on the charging plan. This proposed test program is discussed with the feed sites and the mechanics of reactor use. The test material are currently being studied by the feed sites.

2. The lot charging of small scale tonnages of existing extruded material followed by increasing tonnages until testing is completed and extruded cores are considered standard production. This test program is to be carried out concurrently and essentially independently of the statistical test. HAPC has received sufficient data to start charging the feed sites for the initial lot charging. The data of 100 dingot cores from extruded tubes, 5.1 tons of MCW dingot cores from extruded rods, and 5.1 tons of ingot cores from BBC extruded tubes. The 100 tons of dingot cores from extruded tubes have been examined out-of-reactor including UT-2 testing (10-3% rejected for large grain), and are awaiting canning and reactor charging. Depending upon satisfactory performance, after irradiation of this test lot for four to six months, HAPC would start charging extruded material at the rate of 25 tons per month for six months, 50 to 60 tons per month for another six months, and then to 100 to 150 tons per month until a final order is authorized.

Oil Quenched Cores

Approximately 5 tons of ingot cores fabricated by the process described in the oil-quenched hollow core blanks have been canned and are waiting irradiation. Approximately 100 cores for a statistically designed test to compare oil-quenched hollow core blank treatment with salt-water quenched solid core blank treatment are currently being fabricated by HAPC. Alternate core blanks from

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from the ingot rods are to be oil quenched and water quenched. Following successful irradiation testing of the statistical designed test and initial 6 ton lot charge, the expected rate of charging would be 25 tons per month for six months, 50 to 60 tons per month for an additional six months, and then larger scale tonnages until unlimited use or complete conversion is authorized.

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2. Fuel Assembly Technology

Canning yields by fuel element type for April through July 1961, were as follows:

Fuel Element Type	Percent Yield				Year Date
	April	May	June	July	
8-Inch I&E Natural	88.9	86.8	89.0	91.0	1961
6-Inch I&E Enriched	89.8	89.5	91.2	91.8	1961

Yield improvement during this period is primarily attributed to the use of contour-base aluminum cans and sleeves to accelerate can base heating pre-heating rates, assuring that the AlSi in the can-base is above the primary arrest temperature prior to core insertion, and conversion to a pressurized fuel assembly quench to minimize braze porosity. Other factors included the modification of enriched fuel canning baskets to improve heat transfer, and decreased sensitivity of penetration testers to canwall variations exceeding the minimum specified canwall thickness (20 mils).

Major process improvements planned or introduced during this period are summarized below:

- (a) Installation of pressurized quenching equipment on the production canning lines was completed in August. Results thus far indicate external and internal fuel braze porosity are reduced by factors of about 4 and 2, respectively, through the application of air-pressure (30 psi) to fuel assemblies prior to water quenching. This conforms closely with earlier test data.
- (b) Low-ripple DC-power supplies are being procured to convert the existing AC fuel-closure welders to DC. Delivery of the first unit has been promised in September, and conversion of all welders is

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scheduled for completion by December. Through a series of tests DC-welded closures show about a 44 percent improvement in braze mixing and a significant decrease in weld cracking, voids, and pinholes.

- (c) Tests are also in progress to further optimize brazing cycles and temperatures to achieve maximum bond strength and braze layer integrity. In order to supplement earlier data, which indicated the braze compound layers could be improved by inserting lead plugs in the end of the cores to prevent AlSi from freezing in the bore during submersion in the duplex canning bath, thereby ensuring accelerated and more uniform preheating rates in the lead, more extensive tests on lead-plugged cores are being run in conjunction with the above tests.

3. Irradiation Experience

a) Production Rupture Summary

Twenty (20) natural uranium fuel elements (18 I&E and 2 solid) and 5 enriched I&E fuel elements failed in the Hanford reactors during the period from April 1, 1961, through August 31, 1961. A summary of these ruptures is shown in Table _____.

b) Rupture Trends & Irradiation Experience

A more favorable rupture trend has been established this period. The incidence of ruptures since July 1 is the lowest experienced since conversion to I&E fuel loadings. The overall I&E rupture rate for the first 7 months of this year was 42 per 10^6 fuel elements discharged, as compared with 75 per 10^6 discharged during the same period in CY-1960. Severity of service is over twice that of a year ago.

Fuel assembly oriented ruptures (internal, hole, end and side other) have replaced side hot-spot ruptures as the major category during recent months.

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Fuel assembly oriented ruptures are largely caused by small defects in the weld closures and underlying braze area. As previously reported, these defects are primarily attributed to fuel assembly canning variables and the outgassing of uranium surface hydrogen. Process improvements introduced or planned and control of surface hydrogen on the uranium cores in the near future are expected to essentially alleviate fuel oriented ruptures.

Although specific factors influencing the downward trend in side hot-spot ruptures, associated with fuel misalignment and warp, cannot be readily assessed, major fuel improvements contributing to this trend are: 1) conversion to X-8001 alloy cladding, 2) alteration of fuel core contour angle to prevent potential fuel cocking, 3) revisions of fuel designs to more nearly optimize temperature and coolant flow distribution, and 4) a reduced incidence of fuel elements that warp excessively during irradiation.

A total of twelve (12) ruptures has been experienced to date in natural alloyed dingot-uranium fuel elements undergoing production transition testing, as compared to eight (8) ruptures in the ingot control material. Four of the dingot ruptures were classified as side hot-spot, three as internal or end, two as transverse split core, and three as unclassified. Although the overall performance of dingot uranium is not appreciably worse than ingot uranium, the two recent transverse core failures are of concern because of possible relationship to metal quality. Neither of the ruptured cores show any evidence of transgranular (mid-flat) surface cracks. Detailed examination of the first core failure in Radiochemical Engineering has not revealed the exact failure mechanism. Both transverse and longitudinal cracks were observed in the uranium. At least some of the cracks were caused by water-uranium reaction (See Fig. 3). It is suspected

that if the rupture is metal quality oriented, stress risers in the form of internal cracks may have been of sufficient magnitude to cause failure. Examination of the second rupture is continuing in an attempt to determine the failure mechanism.

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PRODUCTION RUPTURE SUMMARY**DECLASSIFIED**

Date	Reactor	Exposure (MWD/T)	Tube Power at Failure (KW)	Type of Element	Rupture Classification
4-4-61	H	386	1031	Natural I&E	Internal
4-6-61	B	623	1252	Natural I&E	Side Hot Spot
4-7-61	D	745	596	*Natural I&E	Unclassified
4-10-61	KE	422	1637	Enriched I&E	Side Hot Spot
4-19-61	KE	622	779	Natural Solid	Core Split (Long.)
4-21-61	C	382	1064	Natural I&E	Side Other
4-22-61	B	639	1293	Natural I&E	Side Hot Spot
4-24-61	H	542	1102	Natural I&E	Side Hot Spot
4-25-61	KW	744	939	Natural I&E	Internal
4-26-61	H	534	1047	Natural I&E	Unclassified
4-28-61	D	760	631	*Natural I&E	Internal
4-30-61	C	144	1302	Natural I&E	Unclasp
5-1-61	B	491	1262	*Natural I&E	Side Hot Spot
5-2-61	B	450	1167	*Natural I&E	Unclassified
5-17-61	DR	390	1148	Enriched I&E	Base End
6-5-61	C	727	1147	Natural I&E	Side Hot Spot
6-9-61	C	723	585	Natural Solid	Core Split (Long.)
6-9-61	DR	305	977	*Natural I&E	Core Split (Trans.)
6-12-61	F	716	1079	Enriched I&E	Cap End
6-17-61	B	252	1171	*Natural I&E	Base End
6-18-61	H	314	1107	**Enriched I&E	Base End
6-24-61	F	208	1237	*Natural I&E	Cap End
7-1-61	F	419	1172	*Natural I&E	Core Split (Trans.)
8-7-61	H	663	1016	**Enriched I&E	Hole (Base End)
8-23-61	KW	554	1405	Enriched I&E	Side Hot Spot
8-29-61	D	795		**Natural I&E	Hole

* Dingot uranium

** Bumper fuel

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4. E-N Program

Conversion of the H reactor to a full E-N demonstration loading was completed in June. Charging of the second and final cycle of the central "striped" loading of enriched I&E bumper fuel and I&E "N" pieces is scheduled in October. Thus far, two enriched fuel ruptures have been experienced, both of the "base end" fuel quality oriented type.

5. Projection Fuel Program

The D and H reactors are completely loaded with bumper fuel, and charging of the DR reactor is scheduled in October and November. In addition, the enriched fuel loadings in all of the old reactors are being gradually converted to bumper fuel. Irradiation experience thus far has demonstrated that "bumper" fuel performance is better than non-bumper fuel at a 95 percent confidence level and in the order of 2-fold better at a 75 percent confidence level. Of the four bumper fuel ruptures that have been sustained to date, two were attributed to extremely high surface temperatures and uniform corrosion attack, and the other two incurred in the E-N demonstration loading were fuel quality oriented. None of these ruptures could reasonably have been expected to be prevented by bumpers. About 700 tubes of bumper fuel have been discharged at normal goal exposures (800-1150 MWD/T).

In order to reduce the coolant flow resistance offered by the present rectangular solid bumper rail design, an elliptical rail design is being evaluated. Installation of equipment for attaching bumper and self-support rails to fuel elements in the production facility was completed for beneficial use in August.

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Irradiation of self-supported fuel elements in the C reactor 69 ribless zircaloy tube facility is continuing without incident. Sixteen (16) columns of self-supported fuel has been discharged to date at exposures up to 600 MWD/T and 250 columns at exposures from 600 to 1200 MWD/T. Localized and band type corrosion observed between the rails, although not severe, are expected to be minimized by further optimization of fuel and rail design.

6. Overbore Fuel Program

Sufficient quantities of natural self-supported overbore fuel (1.992' O. D. and 0.339' I. D.) have been prepared to expand the original 22-tube C reactor test facility to 62 tubes during August and September. Fuel element yields in the order of 70 percent were achieved, representing a 15-20 percent improvement over earlier experience.

Irradiation of overbore fuel has progressed without incident. Of the eight (8) fringe columns examined that were exposed to 168-775 MWD/T at maximum specific powers of 75 KW/ft., and four central columns exposed from 650 to 825 MWD/T at maximum specific powers of about 105 KW/ft., only normal light to medium corrosion and film patterns were noted on the pieces. Minor hot-spot film patterns were observed on four downstream pieces. These patterns were attributed to high bottom-of-annulus temperatures, which will be reduced by increasing the present self-support rail height from 69 to 72 mils. Fourteen (14) alpha extruded dingot fuel elements and two (2) alpha rolled ingot pieces exposed above 640 MWD/T showed evidence of surface bumping (<10 mils). It is suspected that the larger grain size of the earlier cores received for this program contributed to the bumping.

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7. Nickel Plated Aluminum Clad Fuel Elements

The encouraging in-reactor behavior of 170 chemical nickel plated T&E fuel elements as summarized in NLCO-824 has prompted additional and larger scale testing. Twenty-two hundred (2200) natural uranium fuel elements, including both 0.5 mils and 0.2 mils thick plate, have been prepared for irradiation to approximately one times normal life exposure for comparison of behavior with unplated fuel elements. After plating, the fuel elements were extensively tested to assure plate adhesion, and baked for 5-6 hours at 300° C. Charging is scheduled for early September.

To further test the capabilities of chemically deposited nickel plate it is planned to plate and irradiate approximately 5,000 natural uranium fuel elements; charging is tentatively scheduled for December. These fuel elements will probably be plated in a small-scale facility at HAP0 which has been used for development and testing work for the past 6 months.

AlSi dip-canning of fuel elements in aluminum components plated on all surfaces with chemically-deposited nickel, has been developed sufficiently for irradiation testing. Intermittent work on development of the closure weld has yielded encouraging results, and a small-scale irradiation test will probably be charged this fall.

Cracking of chemically-deposited nickel plate during irradiation as described in NLCO-824 prompted a search for a more ductile plate. According to data developed by Union Carbide Company at Oak Ridge, Tennessee, a significant ductibility increase can be achieved by reducing the phosphorous content of the plate from the normal 8-10 wt. percent to 6 wt. percent or less. A process for the production of low phosphorous

content plate is being developed by the General American Transportation Corporation laboratories at East Chicago, Indiana, and samples of this plate on aluminum have been obtained for testing at HAPC.

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FUEL ELEMENT DEVELOPMENT

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NPR PROGRAM

The N-reactor fuel plant (333 Building) was turned over to Manufacturing personnel on July 1, 1961 and the three-month start-up period is now under way. About forty extrusions have been made by production personnel and the product of these extrusions is being used as material to check out and debug the plant equipment. Most items of equipment seem to be working satisfactorily and debugging is continuing on some few items. A small quantity of inner tubes was extruded in late August as the beginning of actual production and any fuel elements made from these tubes which meet specifications will be considered as reactor quality fuel elements. Plans are to scaleup production and attempt to have the facility operating as smoothly as possible by October 1, 1961.

The fuel fabrication process is firmly established and production specifications have been issued. The process flow sheet is shown in Figure 1. It will be noted that there are a few changes from the flow sheet published in a previous Working Committee report (DPST-61-139).

The pilot plant program has been curtailed somewhat, temporarily, due to the requirement that personnel be available for production plant startup debugging problems. The program of fabrication and irradiation of Zircaloy clad UO_2 fuel elements is continuing but at a fairly low rate. To date two tubes have

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been discharged from KW reactor at about 600 MWD/T and show no unusual conditions. An additional six tubes are under irradiation and for at least the next several months additional quantities will be fabricated and reactor charged as manpower permits. In addition to this program, the pilot plant has fabricated, by prototype methods insofar as possible, all fuel elements being used in the KW loop testing program. Of particular interest on this program is that 1.8% enriched K-single tube fuel elements produced in the pilot plant have been satisfactorily irradiated to 1200, 2000 and 3600 MWD/T without failure.

In addition to the above work, the pilot plant is fabricating moderate quantities (several hundred) full-size fuel elements for use in various KW reactor test programs and tests of such reactor equipment as the charging machine, pickup system, and shipping pallets.

Uranium Billets

The problem of outer clad wrinkling associated with the uranium billet structure was reported in the last Working Committee Report, together with the solution of adding 100-200 ppm iron and 75-140 ppm silicon. Continued experience substantiates this solution to the problem and these levels of impurities are specified as standard production. The only current problems existent in the uranium billet technology are in meeting the carbon and hydrogen specification limits. NLO has generally been unable to meet the 400 ppm carbon limit and in some few instances been unable to meet the 100 ppm hydrogen specification limit.

Coextrusion

As pointed out in the last Working Committee Report, a serious problem of clad bonding had developed. This was determined to be due to leaking of

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the welds of the copper can during preheating and pointed up a definite problem in the billet assembly work. Since this problem was corrected, excellent extrusions with high quality bonds have been made routinely.

The problem associated with leakage of the billets was somewhat difficult to recognize because of the variability of the characteristics of the results. Component tolerances here play an important part, and billets in the past with tightly fitting components have been coextruded with only mild effects upon the bonds. Two solutions to the problem have been worked out. One is to use silicon:bronze (about 3% silicon:copper), a grade of material which welds significantly better than commercially pure copper. A further advantage with this material is the future potential of doing away with the first Zircaloy etching process since the silicon:bronze does not diffuse into the Zircaloy during coextrusion. However, some possible coextrusion lubrication problems have shown indications of being apparent. The other solution has been by changing the welding conditions on the copper and by changing the preshaped contours where the weld bead is made.

Warp Control

Continued experience has indicated that extremely few of the outer tube sections show appreciable warp. It will apparently not be necessary to exercise any straightening of the outer tube sections after beta heat treating. In the case of the inner tube sections, about 50% exceed the warp specification limit of .030 inches TIR. However, nearly all of these can be recovered by reverse bending after the initial beta heat treatment, followed by a second beta heat treatment. It is planned to use this process in production.

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Experiments on 24-inch long N inner fuel elements indicate that hot straightening in a three roll straightener (Hanford design) also produces an element that will meet assembly straightness specifications. Double throw warp following beta heat treatment averaged 0.049 inch, and averaged 0.049 inch. Warp after straightening averages about 0.007 inch with a range of 0.003 to 0.015 inch. Autoclaving following the straightening operation increases the double throw warp an average of 0.005 inch.

End Closure Development

Braze Closure Process

The braze closure process is working satisfactorily although continued development work will be necessary to reduce and eliminate causes of braze porosity. The situation here is somewhat parallel to that with the AlSi braze. As was elsewhere pointed out, fuel elements with the standard zirconium:beryllium braze closure have been irradiated in the KRR test loops to about 3600 M.D/T without failure.

Self Brazing Process

The bond formed at the interface between the Zircaloy cap and the uranium core in the self-brazing process employing a thin copper plate for the bonding medium tends to be brittle. Fracturing in the Cu-U phase commonly occurs due to differential expansion during salt bath beta heat treatment. The procedure was modified to add a flash coat of nickel over the copper on those surfaces of the cap that are to contact the uranium. Destructive tests indicate greatly improved toughness in the interfacial layer.

Gas-pressure Bonded End Closure

The concept of a gas-pressure bonded end closure was described in the last Working Committee report. Briefly, this process entailed re-

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cessing the fuel element end, cathodically etching all mating surfaces, and tungsten welding a ring end cap into place. Subsequently, the piece is gas pressure bonded at pressures of the order of 10,000 PSI and temperatures in the 600-650°C range, the high alpha temperature range for uranium. Additional quantities of samples have been successfully gas pressure bonded since those reported previously. Tensile test samples on the uranium-Zircaloy bond show minimum tensile strengths of at least 40,000 PSI. The strength may be substantially higher than this since the samples tested to date have failed in the grips or the base metal.

All samples processed to date have had carefully machined end closures and mating faces. This is not satisfactory for a production process and if this process is to be satisfactory, chemically milled surfaces will have to work. The fact that bonding between the metals with the pressures and temperatures involved can be achieved has been demonstrated. The problem existing is in designing the end configuration to insure intimate contact between the metal surfaces where bonding is desired. It would appear that by v-notching the uranium and Zircaloy mating faces, plastic flow of the metals will be easier and contact can be established at the pressures being used. This is to be tried.

Fuel Element Supports

The basic design of the inner tube fuel support has been changed. The previous "suitcase handle" design has been replaced with a "buggy spring" design as illustrated in Figure 2. Tests have shown better centering of the inner tube within the outer tube and there is an improvement in the load-deflection characteristics of the support.

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Re-design of the locking device which pins the 8 outer tubes to the inner tube necessitated a change in tooling for attaching this hardware. A new set of tooling was designed for a Weldex Model 10 Spine Tower welder. Approximately one thousand supports were attached to 8 fuel tubes for testing machine tests. These were done on a three-phase 100 KVA Edison welder operating as a single phase welder. Two magnetic force welders have been installed and temporary tooling is in place to permit determination of proper machine settings for support welding.

Failure Studies of Coextruded Fuel Material

IRP Loop Studies

Two 3200 MWD/T exposure coextruded K2O inner fuel tubes, one unalloyed uranium and one U-2 w/o Zr alloy, were defect tested in 300C, 1650 psi water by Coolant Systems Development Operation in the IRP loop. The uranium fuel tube incubated 7-1/2 minutes before failing and the test loop filter accumulated 20R activity just 4-1/4 minutes later. Loop cool-down was started and after five more minutes the loop temperature was reduced to 160C. Examination of the coextruded uranium tube revealed that it had broken into three major pieces and six lesser pieces. The tube had a spiral cleavage from one end to the other (16") and both unbonded end caps had separated. The U-2 w/o Zr alloy fuel tube incubated 10-1/4 hours before failure indications. Eighty-eight minutes later the loop filter activity had reached 20R and shutdown was started. After four minutes the temperature dropped to 200C and fuel corrosion became insignificant and the final filter activity attained 23R. Examination of the failed alloy fuel tube revealed that it had broken into two pieces

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at the cross-section of the defect. There is an order of magnitude difference in the behavior of the fuel tube between the two tests which must be attributed to the alloy oxidation-temperature-time parameters, including irradiation, and the test conditions.

ETR Rupture Tests

The sixth and seventh tests in the series of "push-button" in-reactor rupture tests were performed recently at the ETR. These tests were performed to evaluate the rupture behavior of fuel elements proposed for the Hanford New Production Reactor (NPR). The purpose of the tests was to determine the correlation between the behavior of the "incubation period" of irradiated and unirradiated fuel elements.

The fuel elements used in the last two tests were 18-inch lengths of Zircaloy-2 clad coextruded natural uranium tube. The O.D. and I.D. of the fuel elements were 1.43 and 0.52 inches respectively. The cladding thickness was 0.020 inches. Each test assembly was enclosed in a 1.56-inch I.D. Zircaloy-2 sleeve before it was charged into the reactor. Wall thickness of the Zircaloy-2 sleeves was 0.009 inches. The defect on each of the test fuel elements was located on its outer surface about three inches from one end.

The defected fuel element for the sixth rupture test had been irradiated to 400 MWD/T in a hot water recirculating loop at Hanford prior to its being tested at the ETR. After the test assembly was charged the reactor was brought up to power and operated for four hours and then the defect cap was sheared off. Twelve minutes afterward there was a small

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(1/4 psi) increase in the ΔP across the loop. Thirty-four minutes after the start of the test the loop ΔP had increased by 3/4 psi. Four minutes later, the rupture detection instruments (delayed neutron monitor and gamma scintillators) began rising. The ΔP continued to rise slowly as did the rupture detection instruments, until the reactor was shut down one hour after the start of the test. No sudden, rapid rise in fission product activity in the loop occurred in this test as had occurred in some of the previous tests.

Temperature conditions at the defect during the test were:

Local bulk water temperature	-	304 C
Cladding surface temperature	-	397 C
Cladding-core interface temperature	-	392 C
Core center temperature	-	461 C

The defected fuel element for the seventh test had been irradiated to 1670 MWD/T in a hot water, recirculating loop at Hanford prior to testing at the ETR. After the test assembly was charged into the ETR, the reactor was operated at full power for three hours before the defect cap was sheared off. Nine minutes later, the rupture detecting instruments began rising and the loop ΔP began increasing. In another five minutes the activity had increased to the point that a "slow setback" of the reactor was begun and the reactor was scrammed one minute later.

Test conditions at the defect during this test were the same as during the sixth test of the series.

Although the results of the seven tests have not yet been completely analyzed (the sixth and seventh have not yet been examined in a radiochemistry cell), the following conclusion seems warranted: Irradiated material fails more rapidly, that is, it has a shorter incubation period

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and a more rapid corrosion rate following the incubation period, than unirradiated material of the same geometry under identical conditions. This is probably caused, not by an increased rate of corrosion of the uranium with increased exposure, but by the internal cracking of the uranium which occurs with high exposure, thus exposing a greater surface to the attack of the water.

Irradiation Performance

NER Loop Irradiation of Tubular Elements

The single component tubular coextruded elements designed and fabricated for testing at approximate NER outlet tube conditions in the NER loops have been successfully irradiated to exposures of 1200, 2000, and 3600 MWD/T. The 1200 exposure elements looked good at discharge, except for a "sooty" appearance of the clad. No cause for an effect of this appearance has been determined. Hot extraction of hydrogen from a clad sample of a 1200 MWD/T piece indicated 70 ppm hydrogen, a quantity that is about twice the expected. The 2000 exposure elements also looked good after discharge, except for the lack of the usual black luster of loop tested Zr-2 material and some whiteish areas on the Be-Zr eutectic braze-weld closures that increased in extent in the downstream direction (increasing water temperature). No cause for or effect of the grey clad surface and whiteish closures has been determined. The clad surfaces and closures of the 3600 exposure elements were black (30-50°C lower water temperature than the two lower exposure charges), but the inner bores of the two downstream elements were bumpy.

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The irradiation induced volume increase (swelling) of the elements was determined in the reactor basin by displacement measurements following discharge. The measurements were repeated at least 1 month after discharge to allow for gamma heating decay of the fuel and the known effect on the buoyancy of the elements. There was no significant difference in the data obtained from the two weighings. The calculated swelling of the elements and the fuel is tabulated below:

Exposure WD/M	Tube Position	Vol. Mean Fuel Temp. - C	Element Swelling, %	Fuel Swelling - %
1200	Upstream	430	2.4	2.0
		450	2.4	2.0
	to	460	2.4	2.0
	Downstream	465	2.4	2.0
2000	Upstream	435	2.4	2.0
		450	2.5	2.0
	to	465	2.6	3.1
	Downstream	470	2.5	3.0
3600	Upstream	405	2.4	2.0
		425	2.0	2.0
	to	435	2.4	3.4
	Downstream	440	2.4	3.0

An empirical expression of the fuel swelling observed in the above elements has been derived which relates the swelling to the 1/2 power of the burnup and the temperature in the range 400-475 C. Pre- and post-irradiation densities of the fuel from the downstream element at the 1200 exposure charge revealed a 1.35% increase in volume which compares with the 1.3% determined by displacement measurements in the elements tabulated above.

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Initial sections of the closures of the 1200 and 2000 exposure elements indicate no deficiencies in the Be-Zr eutectic braze closure or in the coextruded bond in the braze heat affected zone. Additional closure examinations of the 2000 exposure elements have revealed some fine radial cracks across the bond in the heat-affected zone. The cracks terminate at the clad, but are propagated across the brittle Be-Zr braze material and terminate at the Zr cap. A cross-section of a downstream element of the 2000 exposure charge revealed some distortion of the inner bore. Buckling of the inner clad at its lower end, under the pressure induced by fuel swelling, is apparently a much more advanced stage of inner bore buckling was noted in the clad on the two downstream elements of the 3600 exposure charge.

Four NPR inner fuel tubes with 0.040" outer clad and 0.025" inner clad were discharged from a KRR loop at 1000 MWD/T. Swelling determinations made by displacement measurements in the reactor basin indicate an average of 0.4% fuel volume increase. These elements are presently entering Radiometallurgy for detailed examination.

Uranium and U-2 w/o Zr KRR Tube/Tube Element Irradiation

Uranium and U-2 w/o Zr KRR elements were exposed to 3200 MWD/T in 270C water. A uranium inner tube failed due to a split clad. The split was the type we have seen before in rod cluster elements and capsule tests. Apparently, the expanding fuel core exerts a large stress on the clad. The clad then fails, with non-uniform elongation causing a longitudinal split. The swelling of the elements was determined by density and dimensional measurements. A summary of the swelling data

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is given in Table I.

The U-2 w/0 Zr swells at a higher rate than Uranium at given core temperature, but the 4000 hr. exposure to the 10,000 hr. test results in swelling.

	B.O.	10,000 hr.	10,000 hr.	10,000 hr.	10,000 hr.	10,000 hr.
						10,000 hr.
A-1 Uranium	0.42	0.0	5.0	13.0	410	10,000 hr.
B-1 U-2 w/0 Zr	0.45	0.0	0.0	12.0	410	10,000 hr.
A-2 Uranium	0.45	0.0	0.0	12.0	410	10,000 hr.
B-2 U-2 w/0 Zr	0.45	0.0	0.0	12.0	410	10,000 hr.

TABLE I

VOLUME CHANGE AND CORE TEMPERATURE DATA FOR KER TUBES EXPOSED TO 3200 MWD / AVERAGE

Irradiation of Zircaloy clad fuel Elements with Brazed Cladures

During the past reporting period one irradiation experiment was performed on fuel elements with brazed claddings. Failure analysis examination was conducted on two other irradiations.

The first rod metallurgical examination was on a fuel element using KER size tube-type elements in the ETR test loop. These elements operated at 32,000 ft. developing 10,000 hr. for the test and failure. The tube was natural uranium clad with a zirconium Zircaloy clad and caps brazed in position with a 100% Fe - 10% Cu - 90% Zr braze alloy.

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At discharge these elements had accumulated 1200 MWD/T (0.14 a/o burn-up) on the outer elements and 825 MWD/T (0.1 a/o burn-up) on the inner elements. Post irradiation warp measurements indicated that the elements warped 0.125 inch toward the higher flux portion of the reactor core with the points of maximum warp corresponding to the flattened end points on the flattened cosine flux curve. Metallurgical examination indicated that the braze had fractured between the core and cap. The inner tube had operated in the beta phase as observed by the columnar grains oriented in the direction of heat flow. The outer tube had operated in the high alpha temperature range above the creep activation temperature. At this temperature and exposure very little evidence of irradiation damage to the uranium was observed. Examination of this experiment is continuing.

Radiometallurgical examinations has been completed on the experiment GER-4-57,50 irradiated in the MTR at a specific power of 17 KW ft² at an exposure of 1100 MWD/T (0.14 a/o burnup). At this exposure the outer elements ruptured. These elements were also natural uranium, Zircaloy clad with end caps brazed in position with the 12 Fe + 4 Be + Zry-2 brazing alloy. Radiometallurgical examination characterized the rupture as follows:

- 1) The element failed by a shear-type mechanism through the cladding at the inner bore. The failure was initiated by cracking of the braze layer between the uranium core and Zircaloy cap. Repeated thermal cycling due to the nine reactor startups and shutdowns coupled with a cracked bond between the cap and core together with the natural swelling of the uranium.

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resulted in the combined forces which caused fracture of the cladding.

- 2) The coextruded bond was completely separated from the Zircaloy in the heat-affected zone above the fracture blister. Bond separation progressed by cracking through both the Zircaloy and along the coextruded bond.
- 3) Swelling was very evident in these elements. The necked-in region of the closure heat-affected zone expanded to a diameter exceeding that of the original element.

A sequel to this test, EKH-4-57,50, was placed in the KBR Loop I, 7, 1961, to irradiate a closure made with the 5 Be + 27.2 Zircaloy alloy. Irradiation conditions were essentially the same as EKH-4-57,50, except the specific power was only 62 KW/ft. These elements will be exposed for six irradiation cycles, then examined in the Hanford Radio-metallurgy facilities.

Irradiation of Elements with Variable Beta Heat Treatment

A production test of N-reactor inner tube fuel with variations in beta heat treating was irradiated in KBR Loop I to approximately 1000 MWD/T. Changes in outer diameter, length, and warp have been determined in the examination of these elements. Outer diameters increased from 0.003 to 0.006 inches in apparent correlation with element location and power; lengths decreased from 0.025 to 0.050 inch in 15-inch length, also in relation to element power; and maximum change in warp was 0.001 inch double throw in the 15-inch length. No evidence of bumping or surface

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distortion was observed even though quenching conditions varied from air cooling to cold water quenching with corresponding variations in structure.

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