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REPORT TO THE WORKING COMMITTEE

FROM HANFORD

by:

M. Lewis - GE
J. E. Minor - BNW
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September 21, 1965

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CURRENT REACTOR PRODUCTION FUELSUranium Core Production DataQuality of Uranium CoresInspection of Virgin Cores

The combined UT-2 and visual bare core reject rate was 0.14 per cent for the period April 1 through August 31, 1965.

Grain Size

The average grain size of cores received during the period February 1 through July 1, 1965 was between A-6 and A-7 on salt-water and induction-oil heat treated cores and A-5 on salt-oil heat treated cores.

Uranium Specifications

There have been no significant changes in uranium specifications during the reporting period.

Fuel Performance

Five normal and eleven enriched AlSi production fuel elements failed during the period April 1 through August 31, 1965. In addition, two natural AlSi overbore elements failed, and three hot-die-sized elements failed. The AlSi failures are summarized in Table I, and the hot-die-size failures are discussed in the section on alternate process development. Nine of the 17 AlSi failures were caused by accelerated side-hot-spot corrosion on non-bumpered fuel, and six were caused by end manufacturing defects (poor end bonding). Three failures occurred in enriched "K" self-support fuels, and all three appear to have been caused by brittle end bonds. The two AlSi overbore failures were also apparently caused by poor end bonding, a problem which has been discussed in previous reports. The general fuel performance has remained essentially unchanged during the past year.

The irradiation-induced warp and external bond count trends are shown in Figures 1 and 2. The warp appears to be reaching the low level experienced in early 1963.

Depleted Uranium

Approximately 12.5 tons of 0.22 per cent U-235 cores were changed in B Reactor during July 1964, and 15 tons were charged in KE Reactor during August 1964. The objective of the program is to produce 25 kilograms of 18 per cent Pu-240 plutonium. To achieve this, average exposures of 1200 MWD/T and 1900 MWD/T are required in the B and KE Reactors, respectively. The performance of the depleted elements has been excellent, and current plans are to irradiate both loads to at least 2500 MWD/T. The dimensional behavior of the depleted elements has been monitored by discharge of selected columns at various exposures. The dimensional data on these columns is summarized below:

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TABLE I

ALSI FUEL ELEMENT FAILURES

<u>Fuel Element Type</u>	<u>Reactor</u>	<u>Per Cent of Goal Exposure</u>	<u>Rupture Classification</u>	<u>Rupture Date</u>
Enriched Self-Support	KE	69	End Manufacturing Defect	4/ 3/65
Natural Non-Bumpered	H	80	Side Hot Spot	4/ 4/65
Enriched Self-Support	KW	75	End Manufacturing Defect	4/ 8/65
Natural Overbore	C	86	End Manufacturing Defect	5/ 8/65
Enriched Non-Bumpered	KE	47	Side Hot Spot	5/ 9/65
Enriched Non-Bumpered	C	41	Side Hot Spot	5/16/65
Natural Non-Bumper	B	81	Side Hot Spot	5/21/65
Natural Bumpered	B	36	Side Mechanical Damage	5/24/65
Enriched Non-Bumpered	C	71	Side Hot Spot	5/26/65
Enriched Non-Bumpered	C	111	Side Hot Spot	6/17/65
Enriched Self-Support	KE	70	Side Manufacturing Defect	6/23/65
Enriched Non-Bumpered	KW	47	Side Hot Spot	7/ 5/65
Natural Non-Bumpered	B	31	Side Mechanical Damage	7/11/65
Natural Non-Bumpered	D	71	Side Hot Spot	7/16/65
Enriched Non-Bumpered	C	98	Side Hot Spot	7/19/65
Enriched Non-Bumpered	KW	57	Not Examined	7/22/65
Natural Overbore	C	105	End Manufacturing Defect	7/28/65
Enriched Bumpered	D	19	End Manufacturing Defect	8/27/65

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BM-5300-074(3-60)
ACCE RICHARD, WASH.

FIGURE 1

FUEL ELEMENT WARP

DEFENSE

FIGURE 2

CANNING DATE VS. EXTERNAL BOND COUNT

UNTRANSFORMED

Average Bond Count Log (x+1)

Reactor	B		KE		
Exposure (MWD/T)	775	1630	775	1457	1944
ΔOD_1 (mils)	1.8	4.4	-1.7	5.7	4.25
ΔOD_2 (mils)	3.8	6.31	-0.5	7.3	4.77
ΔOD_3 (mils)	4.8	8.37	-3.3	2.0	8.01
Warp (mils)	6.0	2.21	10.6	2.7	3.07
TFC (mils)	11.0	10.1	12.0	11.3	9.3

In both the B and the K loads, the warp appears to have maximized at a relatively low exposure and then declined as exposure increased. This behavior is in keeping with the mathematical models that have been developed to predict irradiation effects. The corrosion on depleted elements has been slight with some pitting less than 10 mils deep.

Projection Fuel Program

The bridge-type self-support rails for fuel loaded in ribless tubes have been replaced with "arch" self-supports. As reported earlier, the arch self-support has the advantage of lower cost, a better load-deflection curve, lower hydraulic resistance, and only one weld is required on each rail. Conversion from the bridge to the arch self-support was completed in July 1965.

Induction Heat Treating

A total of 157 tons of induction heat treated cores have been charged in the small reactors. The performance has been satisfactory, and the post-irradiation data on five columns shows warp at about the same level as experienced with salt-water and later salt-oil material and O.D. growths which are less than that observed on either salt-oil or salt-water heat treated cores. A comparison of the radial growth index on salt-oil and induction-oil heat treated cores is shown in Figure 3. These data support the irradiation experience.

ALSi Process Development

Cap End Bonding Problem

Until January 1965, the external bond testers were set to test the external bond to within approximately one-fourth inch of the cap end of the core. In December 1964, it was discovered that some elements had fractured bonds in this untested area, and the testers were adjusted in January 1965 to test closer to the core end. In July, circuitry changes were made to permit the external bond test to scan past the core end and into the cap-to-can wall braze area.

The K5N fuel elements were found to be more susceptible to fractured end bonds than other models, and after January the external bond reject rates were excessive. The end-bond structure was found to be mostly the easily fractured USi_2 compound rather than the desirable UAL_3 compound. The bond does not crack

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FIGURE 3

RADIAL GROWTH INDEX G-2
SALT-OIL AND INDUCTION-OIL HEAT TREATED CORES

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until after the element has been thermally shocked during cap welding. The base end also has the USi_2 structure but base welding does not fracture the bond.

During part of April and all of May, the K5N cores were canned using a low lead temperature which reduced the external bond reject on virgin cores from eight to twelve per cent to approximately four per cent. However, the bond reject rate was still not as low as desired.

Small-scale canning tests in May indicated that using low silicon (approximately seven per cent) in the duplex AlSi (rather than the usual nine per cent silicon) produces K5N elements with a large percentage of the UAL_3 structure on the core ends. Approximately 40,000 K5N's have been canned in July and August using low silicon (6.7 to 7.7 per cent) in the duplex bath AlSi. About 53,000 control elements using normal silicon conditions (8.5 to 9.5 per cent) in the duplex bath AlSi were also canned. The low silicon test was shifted from line to line and the silicon concentration within the test and normal conditions was adjusted for various silicon levels. However, at silicon concentrations below 7.5 per cent, the external bond reject rate is high due to large voids between the UAL_3 compound layer and the AlSi which seem to be coated with a dross-like structure. Above 9.1 per cent silicon the bond reject rate is again high, primarily due to brittle bonds. Large-scale canning tests with the duplex bath silicon concentration in the 7.5 to 8.5 per cent range are planned.

Thoria Program

Approximately 200 tons of thoria target elements have been charged in the Hanford reactors. The thoria has been canned in two campaigns, one from November through April, and the other June through July 1965. Ten failures occurred in thoria elements during the reporting period, bringing the total number of failures to 21. The ten failures are summarized in Table II. All of the failures occurred in material canned during the first thoria campaign. The failures examined have all been apparently caused by water entry through the weld closure. In late February, tightened inspection of welds was instituted, and 20 of the 21 failures occurred in the ninety tons of thoria elements which were fabricated prior to tightened inspection. The improvements in inspection consisted of helium leak testing, radiography, and improved visual inspection.

The principal problems in production continue to be in welding, compaction, and moisture pickup. Welding yields during the past campaign ranged from 40 to 95 per cent. Test data indicates that a void area beneath the cap will cause poor weld yields due to blowouts, and that this can probably be offset by use of a chill ring to remove heat from the element during welding or by putting the maximum possible amount of thoria in a can. The adsorption of moisture presented problems during the canning of thoria. On 64 lots sampled after canning, the average loss-on-ignition analysis was 0.101 per cent with 3 σ limits of 0.011 to 0.191 per cent. The average increase in LOI during canning was 0.028 per cent. The increase in LOI was determined to be due to moisture.

The past run was conducted largely with Mallinckrodt Chemical Works' oxide, and the compactability was variable. Accordingly, the target density was established as 7.0 ± 0.2 g/cc in the cans, even though a fair number of lots

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TABLE II
THORIA TARGET ELEMENT FAILURES

<u>Failure Date</u>	<u>Residence Time (Days)</u>	<u>Reactor</u>	<u>Hanford Lot No.</u>	<u>Element Density (g/cc)</u>	<u>Oxide⁽¹⁾ Lot No.</u>	<u>Oxide Analyses (Vendor-Supplied Sample)</u>		
						<u>Vendor LOI (%)</u>	<u>Hanford LOI (%)</u>	<u>Hanford Offgas (cc/g)</u>
5/ 3/65	64	KW		7.4				
5/ 3/65	64	KW	TK-6940	7.4	N30		0.04	0.568
5/ 3/65	64	KW	TK-5040	7.4	N18	0.03	0.06	0.192
5/ 3/65	64	KW						
5/12/65	139	C	TC-1550	7.5	N7 T208	0.01 0.05	0.05 0.02	0.133
5/12/65	139	C	TC-1650	7.5	N7 T209	0.01 0.04	0.05 0.03	0.133
5/14/65	141	C	TP-0210	7.1	T130 T126	0.02 0.07	0.07 0.04	0.100 0.202
6/23/65	181	C	TC-1250	7.5	N5	0.03	0.05	0.228
6/26/65	54	KW	TK-9240	7.4	N58	0.03	0.05	0.126
7/28/65	216	C	TC-1250	7.5	N5	0.03	0.05	0.228

(1) Lot Numbers: MCW lots have letter "T"
NFS lots have letter "N"

were capable of being compacted to considerably higher densities. The oxide received was characterized by variable particle density, surface area, offgas and LOI. Table III presents analytical data characterizing the MCW oxide.

Evaluation of GE-NMPO Thoria

Three-hundred pounds of thoria were received which had been produced by the GE-NMPO extrusion process. The Hanford analyses on this material were:

	Surface Area <u>M²/g</u>	Offgas <u>1200 C cc/g</u>	Dissolution <u>%</u>
1/8 inch pellets	0.78	0.029	98.6
- 6 + 10 mesh	0.41	0.099	98.2
- 60 +100 mesh	0.44	0.093	99.4
-200 +325 mesh	0.66	0.156	99.6

Several compaction tests were conducted in production equipment, and the highest density achieved was 7.23 g/cc. This was reached with the following particle size mix: 60 per cent -6 +8 mesh, 20 per cent -35 +48 mesh, and 20 per cent -325 mesh particles. The reason for the relatively low compaction density is apparent from the results of mercury penetrometer tests. Figure 4 shows the results of mercury penetrometer testing of arc fused, sol-gel, and extruded thoria. At low mercury pressures, the sol-gel and arc fused thoria both show particle densities considerably higher than the extruded. However, at pressures of 300 to 1000 psi, the density of the extruded material increases rapidly and reaches that of arc fused thoria. The data show that the extruded thoria has an effective particle density of around eight g/cc, has a void volume of around 20 per cent, and that 90 per cent of the voids are in the 0.3 to 0.6 micron size range.

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TABLE III

ANALYTICAL DATA ON MCW THORIA

Lot	Tap Density ⁽¹⁾ g/cc	Compaction ⁽²⁾ Density (g/cc)	Particle Density (g/cc) ⁽³⁾	BET Surface Area (m ² /g)	Offgas Volume 1200 C cc/g	Loss-on-Ignition %
376	7.0	6.82	8.87	3.12	0.087	0.04
395	6.9	6.66	8.64	4.34	0.043	0.08
406	7.0	6.94	8.74	4.99	0.020	--
539	7.2	7.10	8.75	3.60	0.237	0.07
563	7.3	7.16	8.85	3.34	--	0.10
577	7.2	7.10	9.10	2.39	--	0.09
580	7.3	7.21	9.18	3.24	--	0.09
583	7.2	7.10	8.96	3.98	0.751	0.14
585	7.3	7.12	--	3.15	0.144	0.06
598	7.1	7.05	--	4.04	0.214	0.09
608	7.1	6.66	--	2.83	0.195	0.10
612	7.3	7.28	--	3.39	0.195	0.06
616	7.1	7.50	--	1.89	0.245	0.11
638	7.2	7.10	--	3.88	0.272	0.12
644	7.4	7.18	9.08	2.91	0.753	0.09
647	7.5	7.40	9.33	1.69	0.114	0.05
658	7.7	7.67	--	1.61	0.380	0.07
664	7.7	7.60	9.64	1.14	0.138	0.06
667	7.5	7.25	--	3.15	0.325	0.09
683	7.8	7.62	--	1.09	0.079	0.10
686	7.4	7.48	9.42	1.27	0.162	0.06
693	7.6	7.61	--	2.24	0.257	0.08

(1) Mechanical tapper

(2) Hanford production vibrator

(3) Hg penetrometer - 3000 psi

FIGURE 4 - THORIA PARTICLE DENSITY

N-REACTOR FUELSProduction Status

Second quarter Shop yields for coextrusion continued to improve over first quarter yields. Over-all first quarter CY-1965 yields were 66.81 per cent, while those for the first seven months of this year were 69.65 per cent. Inner fuel yield has continually improved during the year moving from 68.1 per cent at the beginning of the year to a current level of 78.7 per cent.

With the exception of a yield loss during startup of the spike enriched fuel campaign, no unusual fabrication problems were encountered during the second quarter. This loss was caused by normal plant "shakedown" at the start of a new fuel geometry manufacture.

Inner fuel warp rejects continued to decrease during the second quarter.

Improved Material Utilization

Both zircaloy and uranium utilization can be improved by increasing the total length of the coextruded fuel tube without a proportional increase in the length of the end defect. This increase can be accomplished by increasing both the billet diameter and billet length. The work to date has been on the standard alloy in the N-outer fuel geometry, and it now appears that a three per cent gain in uranium and zircaloy cladding utilization can be attained by converting to the following billet design.

	<u>Uranium Dimensions</u>				<u>Average Estimated Extrusion Pressure</u>
	<u>O.D.</u>	<u>I.D.</u>	<u>Length</u>	<u>Weight</u>	
New Design	7.405	2.890	24-1/2	620 lbs.	2700
Present Design	6.950	2.800	19-3/4	430 lbs.	

Four test coextrusions have been made with the larger diameter billets to confirm the calculated extrusion pressures and no problems were encountered. The added length will be accommodated by permitting 2-1/2 inches of the three-inch thick steel dummy block to be in the unsupported end of the liner prior to upset. Remaining length increase will be obtained by using the two-step coextrusion process. In the two-step coextrusion process, a graphite follower block is not inserted between the billet and the steel dummy block during the first stage of the coextrusion. The coextrusion is stopped while the die cone is still filled with materials from the rear of the billet assembly. At this point, the mandrel is retracted and the graphite cutoff is inserted behind the die cone and the coextrusion completed in the second step. All of the usable fuel material is coextruded through the die in the first step. Fourteen N-outer coextrusions have been completed by this two-step technique and no problems were encountered.

N-Inner Fuel Warp

A continuous and occasionally extreme fuel loss results from excessive warp after heat treatment of N-inner fuels. Recent studies have indicated that some warp may be closely related to the heat treat process itself. Recent

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findings indicate that inner fuel elements are not heated uniformly across the fuel diameter. Differences as great as 15 C have been found from one side of a fuel to the other. Two possible causes of this condition are being studied with the following results:

1. The immersed electrode bath heating system has been shown to produce currents between electrodes which are connected to separate, isolated transformer secondaries which is contrary to the expectation that currents would only be found between electrode pairs on the same transformer secondary. Voltage differences as high as 2.1 volts A.C. have been detected between ungrounded fuels and ground. Voltage measurements between unpaired electrodes ranged from 4.2 to 14 volts A.C. Current through a 12 gauge copper wire shorted across unpaired electrodes ranged from 130 to 300 amps. This study has shown a possibility that elements supported from a full heat treat hanger may become part of a circuit in the bath and carry relatively large currents.
2. Temperature measurements during heating and salt bath quench have shown a difference in heating and cooling rates between the fuel surfaces nearest and furthest from the heat treat hanger. Fuels which have been marked to orient them with the hanger have shown some preferential warping in this direction. It has been found that the more extreme temperature difference, 15 C, was experienced during cooling in the salt bath quench. The study of the various relationships between heat treating and warp is continuing.

N-REACTOR FUEL DEVELOPMENT

Process Development

Uranium Fabrication

Upset-Forging of Uranium Billets

Upset-forging of billets is a candidate fabrication route. Some development work has been completed and initial coextrusion testing of forged product is planned for September 1965.

The process is based on the primary extrusion of ingots to a tube of less wall thickness than the standard process. The extrusion outer diameter is approximately 0.350 inches less and the inner diameter approximately 0.300 inches greater than the dimensions of the standard extrusion tube.

Following extrusion, the tube is cut into (raw) billet blank lengths, reheated to extrusion temperature, and upset in the press between dummy blocks to finished dimensions. Early test results show that acceptable billet surfaces are produced from raw billet stock provided no surface breaks occur in the raw stock. The tests have also shown that dimensional control of the forged billet is good, with the exception of center wall thickness variation. It has been found that the center wall thickness variation of the finished billet will be approximately the same as in the raw billet. Correction of wall variation by boring is feasible since

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variation in I.D. is important and only metal symmetry around the I.D. is required for acceptable forging stock.

An upset-forge development campaign was concluded during July at RMI on both N-outer and Mark D (Simulated Driver Tube) geometries for coextrusion in the 8.1 inch liner. All N-outer billets were extruded using a tipped mandrel starting from 13-inch outer diameter ingots of standard production alloy. Mark D billets were extruded over a straight mandrel from 13-inch outer diameter ingots in higher additive alloys. Wall thickness variation and surface defects were corrected by machining the raw billet stock prior to upset-forging. Defects which were removed by drilling or grinding were completely filled during upset providing the residual salt from the preheat bath was removed before upsetting. In spite of defect corrections to the raw billets, uranium utilization was high. Two N-outer ingots produced four billets each with an average uranium utilization, based on finished billet weight divided by casting charge weight, of 73.6 per cent. This may be compared with an approximate 50 per cent yield in the standard production process. Average uranium utilization for three Mark D ingots was 58.6 per cent. This lower figure resulted from long end crops of insufficient length for a fourth billet. In one case, the front end crop was approximately two inches short of the additional billet. This problem is correctable by adjustment of casting charge weight. Average wall thickness variation for the finished billets was 0.027 inches and 0.016 inches for the N-outer and Mark D geometries, respectively. The specified maximum for machined billets is 0.030 inches.

A large campaign to produce 50 N-outer pre-shaped billets by this process is planned for late September 1965. It is expected that more detailed dimensional and yield data will be obtained.

Alternate Uranium Composition-Fuel Alloy Evaluation

Both operating conditions for driver fuel elements of a single-tube design and extended goal exposure of the current tube-in-tube design require development of a fuel material that is more swelling resistant than the current 150 ppm Fe plus 100 ppm Si alloy composition. Single tubes having uranium compositions shown in Table IV have been both beta and gamma-heat treated by the salt-salt process and partial results of evaluation tests have been previously reported on: (1) examination of microstructure and bond zone, (2) hot hardness tests at 300 and 400 C, (3) determination of recrystallization and hardness recovery temperatures after working to 21 per cent reduction in area and (4) stress cycling tests at 400 and 500 C. Further data on the above tests are reported with new results on: (1) determination of low temperature tensile ductility transition, and (2) defect corrosion behavior in water at 300 C under 1400 psi.

As a basis for comparing the materials, the volume per cent of the various intermetallic compounds and their composition was calculated and the totals are shown in Table IV. Also shown in Table IV is the total atomic per cent additive that could be present if a metastable solid solution existed during irradiation with no intermetallic compound formation. These comparisons show that if keying of dislocations by second phase particles is

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TABLE IV

EVALUATION OF URANIUM BASE FUEL MATERIALS

<u>Alloy Composition</u>			<u>Volume Per Cent Compound</u>	<u>Total a/o Addition</u>	<u>Hot Hardness (Kg/mm²*)</u>				<u>Recrystallization Temperature**</u>		<u>Hardness Recovery Temperature***</u>	
<u>Fe (ppm)</u>	<u>Si (ppm)</u>	<u>Al (ppm)</u>			<u>300 C</u>		<u>400 C</u>		<u>Beta</u>	<u>Gamma</u>	<u>Beta</u>	<u>Gamma</u>
173	109	63	0.73	.220	94.6	93.1	60.3	60.8	474 C	480 C	445 C	445 C
269	345	45	1.87	.449	98.5	101.7	70.8	75.1	505 C	505 C	465 C	470 C
250	248	247	1.69	.535	94.8	96.9	71.7	75.0	505 C	505 C	455 C	465 C
395	79	823	2.31	.955	105.0	101.2	68.8	74.6	500 C	525 C	450 C	465 C
362	347	700	2.89	1.062	129.0	123.5	85.9	90.6	515 C	525 C	485 C	510 C

* Average of tests run at 400, 600, and 800 pound loads, four hours loading, 10 mm WC ball.

** Temperature for 50 per cent recrystallization in 20 hour anneals of 21.4 per cent cold worked material.

*** Temperature for 50 per cent decrease in hardness in 20 hour anneals of 21.4 per cent cold worked material.

primarily responsible for elevated temperature deformation behavior, recovery processes, and irradiation performance, the alloys would fall into roughly three groups. If the metastable solid solution assumption is made, the alloys again fall into three groups but with more separation between the groups. The hot hardness and recovery data also shown in Table IV generally support this grouping of alloys; i.e., a change from the standard composition vs. the next three alloys vs. the fifth alloy. Only slight changes are noted due to heat treating conditions. The recently completed stress cycling data given in Table V show that only those alloys containing aluminum have greater resistance to plastic deformation at 400 and 500 C.

Since the last report, tensile ductility was measured from room temperature to 250 C for these alloys. All of the materials exhibited a marked change in ductility from an average of four to five per cent elongation at room temperature to an average of 20 to 25 per cent elongation at slightly elevated temperature. The temperature corresponding to the midpoint of this ductility transition was determined from the ductility curves and is shown in Table VI. It is seen that there is a slight increase in transition temperature with higher alloy additions and, in general, a higher ductility transition is observed in the gamma heat treated condition.

Corrosion tests of the alloys have been conducted in both heat treated conditions using short, deliberately defected lengths of full tube cross section. A window autoclave, large enough to test this size fuel, was installed and photographic equipment obtained for following the test by time-lapse photography. The tests are expected to be completed during the next reporting period.

Uranium Alloy Capsule Irradiations

Studies are in progress to evaluate uranium base alloys in terms of composition, fuel fabrication history, corrosion behavior, and irradiation performance. Forty irradiation test capsules and five temperature monitoring capsules were charged in five process tubes in a production reactor. The irradiation includes 120 specimens of 17 different uranium base alloys containing small additions of Zr, Nb, Mo, Al, Si, Fe, P, and C in various conditions of heat treatment. The compositions and heat treatments have been reported in previous Hanford reports to the Working Committee. All thermocouple systems have continued to operate satisfactorily. Maximum uranium temperatures in each of the five columns of capsules varied from 600 C in the peak flux central capsules to 450 C in the capsules at each end of the columns. One column of eight specimen capsules and one thermocouple capsule was discharged at an exposure of 2300 MWD/T for the center capsule and 1500 MWD/T for the end capsules. Examination of the fuel in these capsules is scheduled to start at the end of August. The remaining four columns of capsules have now reached a maximum exposure of approximately 3300 MWD/T. Goal exposure for these capsules is 4000 MWD/T.

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TABLE V
ELEVATED TEMPERATURE STRESS
CYCLING TESTS OF URANIUM ALLOYS

<u>Alloy (ppm)</u>	<u>Heat Treatment</u>	<u>Cycles</u>	<u>Total Strain</u> (Per Cent in 1.125 Inch Gage)	
			<u>400 C (1)</u>	<u>500 C (2)</u>
1. U + 150 Fe + 100 Si	β HT	500	.44	1.51
		1000	.62	1.51
		5000	.53	2.22
	γ HT	500	.71	.53
		1000	.98	1.69
		5000	4.62	18.30 (3)
2. U + 250 Fe + 250 Si + 250 Al	β HT	500	.27	.09
		1000	.44	.27
		5000	1.60	.71
	γ HT	500	.09	.09
		1000	.18	.35
		5000	.18	.53
3. U + 350 Fe + 800 Al	β HT	500	0	.35
		1000	.18	.35
		5000	.27	1.16
	γ HT	500	.27	.09
		1000	.35	.62
		5000	.18	.44
4. U + 250 Fe + 350 Si	β HT	500	.27	.89
		1000	1.51	2.66
		5000	.27	1.24
	γ HT	500	.35	.27
		1000	2.04	.62
		5000	.18	1.51
5. U + 350 Fe + 350 Si + 800 Al	β HT	500	.27	.27
		1000	.27	.18
		5000	.18	.62
	γ HT	500	.0	.18
		1000	.0	.09
		5000	.18	.35

- (1) Stress level 30,000 psi - cyclic stress $\pm$ 3000 psi - 0.1 inch/min. cross head speed.
 (2) Stress level 15,000 psi - cyclic stress $\pm$ 3000 psi - 0.1 inch/min. cross head speed.
 (3) Failed at 3518 cycles.

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TABLE VITENSILE DUCTILITY TRANSITION
TEMPERATURE OF URANIUM ALLOYS

<u>Alloy Composition (ppm)</u>			<u>Ductility Transition Temperature °C</u>	
<u>Fe</u>	<u>Si</u>	<u>Al</u>	<u>Beta Treated</u>	<u>Gamma Treated</u>
173	109	63	85	105
269	345	45	90	110
250	248	247	100	90
395	79	823	80	135
362	347	700	100	95

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Irradiation ExperienceN-Fuel Failure Analysis

During this current report period, "N" Reactor incurred three fuel element failures. These failures have been designated as No. 2, 3, and 4; failure No. 1 was caused by a defective end cap and it occurred on July 6, 1964. Information on the last three failures is summarized below:

<u>Failure No.</u>	<u>Failure Date</u>	<u>Remarks</u>
2	5/15/65	The site of failure was the inner clad of an outer element and was immediately adjacent to the contact area of a buggy-spring support on an inner element. The fuel column was first-load material in the fringe zone and had attained an exposure of only approximately 400 MWD/T.
3	6/ 6/65	The site of failure was again the inner clad of an outer component immediately adjacent to the contact area of a buggy-spring support. The column was first-load material in the fringe zone and had attained an exposure of approximately 800 MWD/T. There was a remarkable resemblance in appearance to failure No. 2.
4	6/15/65	There was extensive damage and fuel corrosion on the inner bore along with fuel cracking and clad tearing on the outer surface of a co-product driver element. The site of failure was immediately next to a projection weld on a locking clip located on the inner bore of the driver. The column was charged in the central zone as production test material and had attained about 1400 MWD/T exposure after five months' residence.

Radiometallurgical examination and analysis of the three failures have been completed. Failure No. 2 and No. 3 were caused by fretting corrosion of the inner clad of both outer elements by entrapped foreign material on the upstream side of the inner component buggy spring type supports. In both of these failures there are fretted areas on the supports. The fretting member in failure No. 2 was not found, but it is concluded to have been a wire-shaped piece. In failure No. 3, the fretting member was mechanically locked in the buggy spring support and was recovered and identified as a wire-shaped piece of chromium plating. The fretting members are both thought to have come from chromium-plated valve parts in the primary loop system.

Failure No. 4 was caused by a corrosion penetration of the inner clad of the driver element in the crevice under the locking clip immediately adjacent

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to but not part of either the projection weld or the heat-affected zone surrounding the weldment. The penetration is pear-shaped and is only 0.010 to 0.020 inches in diameter on the coolant side of the 25-mil thick clad; it is nearly 0.070 inches in diameter at its maximum dimension toward the fuel side of the clad. This pear-shaped cavity is full of zirconium corrosion product and the shape suggests a response of the corrosion reaction in relation to the temperature gradient across the clad. The most plausible explanation for the failure is the entrapment of fluoride etchant in a small defect and subsequent in-reactor corrosion of the defect to result in the pear-shaped penetration of the clad. The origin of a prior defect in the clad to permit this explanation cannot be adequately explained.

Fluted Single Tube Irradiation

A fuel concept employing a fluted outer surface as a means for extending the total exposure capability of metallic uranium fuel has been the subject of an irradiation test in the ETR. Two elements of N-single tube size containing eight flutes about 0.1 inch high were exposed. Failure of one element was disclosed in an earlier report to have terminated the test after an exposure of 2500 MWD/T. Recent examination of the irradiated fuel indicated that the design performed as intended in accommodating swelling. However, unexpectedly large amounts of swelling occurred in some portions of the fuel; near each end of the element, a volume increase of nearly ten per cent occurred. The cause for this localized, extreme swelling near the ends of the element has not been determined.

Uranium Swelling in N-Reactor

Nearly one-hundred columns of fuel elements, which were charged in the first loading of N-Reactor, contain some measured and weighed elements. These columns, called monitor columns, are to be discharged at progressively higher exposures of the first load to evaluate fuel element performance indices. One of the most important indices that will be evaluated is swelling of uranium. This index will provide an important base for comparison of the present uranium alloy of 150 ppm Fe plus 100 ppm Si with newer uranium alloys.

Five monitor columns have been discharged at this time. All of these columns had reached tube-average exposures in the range from 1554 to 1631 MWD/T. Uranium swelling values were measured on twenty-three (23) outer elements and nineteen (19) inner elements from these columns. These N-Reactor swelling values are compared with values that were obtained on fifty-five (55) outer and inners irradiated as fuel element assemblies in the KER loops in KE Reactor.

In advance of presenting a comparison of fuel element irradiations in two reactors, the following differences in irradiation conditions in each reactor should be kept in mind.

- 1) The entire irradiation history of N-Reactor fuel elements has been in a closed pressurized loop with inlet water temperatures above

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347 F (175 C). The KER data are from four tests where equivalent inlet coolant temperatures were maintained for 63 to 86 per cent of the time of the irradiation cycle; for the remaining time of the irradiation cycle, inlet temperatures were 68 F (20 C) and the loops were not pressurized.

- 2) KER test columns were shorter columns; they were charged with either thirteen or fourteen assemblies (nominal 24-inch lengths).

N-Reactor fuel columns contain the equivalent of 18, 24-inch long assemblies (in various combinations of 24, 18, and 12-inch lengths). Thus, ends-of-the-column elements operate at markedly different specific power.

- 3) N-Reactor is under-moderated in comparison with KE Reactor and the ratio of power and burnup in an outer vs. an inner element is quite similar in N-Reactor and quite different in KE Reactor.
- 4) The test irradiations in KE Reactor were conducted under full-power operating conditions for the period of the exposure. N-Reactor fuel elements have had a much longer in-reactor residence due to the fact that extensive testing of the reactor was necessary as power was gradually increased to design power.
- 5) Swelling results that have been reported for KER tests have been subjected to review, correction, and analysis that time has not yet permitted for the N-Reactor data in this preliminary report. Some few corrections in these N-Reactor swelling data can be expected as both average weight values and the individual values, from which average values were computed, are subjected to statistical analysis and interpretation.

Figure 5 shows two nearly parallel curves that define the range of swelling values for outer uranium in KER tests when plotted against an arbitrarily chosen index of the product of exposure and volume mean uranium temperatures (i.e., $MWD/T \times ^\circ K$). Superimposed on that plot of KER swelling are the 23 swelling values obtained in N-Reactor. In spite of the fact that $^\circ K$, the volume mean uranium temperature, has been calculated on a different basis (neutron flux and ends-of-the-column effect) there is reasonable agreement between these data at index values greater than 1.05 ($MWD/T \times ^\circ K$) but noticeably more swelling in N-Reactor results at lower index values.

Figures 6, 7, 8, and 9 show the same 23 outer-uranium swelling values plotted by position of each swelling value in the fuel column. These four figures each show two different plots. The top plot shows measured swelling ($\% \Delta V$) and exposure (MWD/T) vs. fuel element position. The bottom plot shows measured swelling and calculated swelling vs. fuel element position. Calculated swelling values have been computed from an equation developed by Battelle Northwest Laboratory using swelling values measured in KER irradiated outer uranium and expressed as a function of burnup, volume mean uranium temperatures, and system restraint (i.e., the sum of coolant pressure plus clad strength).

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FIGURE 5

COMPARISON OF NPR OUTER URANIUM SWELLING WITH
OUTER URANIUM SWELLING IN KER TESTS

RI-REA-2526

FIGURE 6

OUTER URANIUM SWELLING IN TUBE 1656TOP: VS EXPOSUREBOTTOM: VS CALCULATED SWELLING

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FIGURE 7

OUTER URANIUM SWELLING IN TUBE 1659

TOP: VS EXPOSURE

BOTTOM: VS CALCULATED SWELLING

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FIGURE 8

OUTER URANIUM SWELLING IN TUBE 2753

TOP: VS EXPOSURE

BOTTOM: VS CALCULATED SWELLING

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FIGURE 9

OUTER URANIUM SWELLING IN TUBE 2758

TOP: VS EXPOSURE

BOTTOM: VS CALCULATED SWELLING

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The top plots in Figures 6 through 9 show the expected correlation of swelling with exposure when plotted for each element's position in the fuel column because this plot serves to eliminate the effect of variation in volume mean uranium temperature at a fixed value for system restraint. The bottom plots in Figures 6 through 9 show the expected correlation of calculated vs. measured swelling in light of the variation between the measured swelling in N-Reactor and the measured swelling in K Reactor (Figure 5).

Figure 10 shows two nearly parallel lines which represent the range of inner uranium swelling values from KER tests vs. the arbitrary index of exposure times volume mean uranium temperature. Again, values of inner uranium swelling for 19 elements irradiated in N-Reactor are superimposed upon the KER data. There is much more scatter in these inner swelling values at any given product of exposure times volume mean uranium temperature than was observed with the outer uranium swelling values. The magnitude of these unpredicted large swelling values will be subjected to extensive review, but equal effort will be devoted to the immediate weighing of other inner fuel elements to assess the probability that these values are correct and take remedial action. Of particular concern is the limiting strain of the outer cladding of the inner fuel element. The design limit for outer clad strain is believed to be one to two per cent (i.e., one to two per cent total strain) based on past experiences where selective strain (i.e., necking) has been shown to occur at larger values of total strain.

Figures 11, 12, and 13 show two different plots. The upper plots show inner uranium swelling and exposure as a function of the position of the element within the column and the lower plots show measured inner uranium swelling and calculated inner uranium swelling as a function of position within the column. Both sets of plots show more variation than was found in those same plots for outer uranium and the variation was expected from Figure 10.

The next monitor columns that will be discharged will have tube average exposures at approximately 2000 MWD/T. Immediate attention will be given to weighing inner fuel elements to determine uranium swelling and the elements will also be measured so dimensional changes can be used to corroborate the swelling values obtained.

CURRENT REACTOR FUEL DEVELOPMENT

Alternate Process Development

Production Experience

There have been no major problems in producing hot-die-size elements in the pilot plant. The design capability of 5000 pieces per month has been realized, and throughputs of 1200 to 1500 pieces per week have been demonstrated. The yields are around 85 per cent, with the largest single reject category being due to marred surfaces, a problem that will improve through further automation. A serious welding problem was solved by increasing the size of the end caps slightly, and welding yields are now better than 95 per cent.

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FIGURE 10

COMPARISON OF NPR INNER URANIUM SWELLING
WITH INNER URANIUM SWELLING IN KER TESTS

FIGURE 11

INNER URANIUM SWELLING IN TUBE 2753TOP: VS EXPOSUREBOTTOM: VS CALCULATED SWELLING

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FIGURE 12

INNER URANIUM SWELLING IN TUBE 2758

TOP: VS EXPOSURE

BOTTOM: VS CALCULATED SWELLING

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FIGURE 13

INNER URANIUM SWELLING IN TUBE 2851TOP: VS EXPOSUREBOTTOM: VS CALCULATED SWELLING

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Irradiation Experience

As of September 8, 1965, a total of 7,826 hot-die-sized elements have been charged and 2,366 discharged, leaving 5,460 pieces in the reactors. The breakdown by model is:

<u>Model</u>	<u>No. Charged</u>	<u>No. Discharged</u>	<u>No. in-Reactor</u>
C6AN	1,244	190	1,054
O3AN	64	64	0
K5AN	4,066	0	4,066
C5AN/C5AE	2,044	2,044	0
O3AE	<u>408</u>	<u>68</u>	<u>340</u>
Total	7,826	2,366	5,460

Five failures have occurred in hot-die-sized fuel and these are summarized in Table VII. The two enriched failures occurred in a production test and are discussed in the section on the development program. In the overbore failures, the initial failure exhibited detachment of approximately 1/4 of the cap from the core due to swelling of uranium corrosion products. The cap appeared to be well bonded to the core over the remainder of its area. A dark pit on the I.D. weld appeared to be the path of water entry, based on underwater examination. The failures which occurred on August 21 and September 12, 1965 were similar to each other in that the point of water entry appeared to be the point of attachment of a support rail. The support rail was flattened for the August 21, 1965 failure and missing on the September 12, 1965 failure. Other rails on the ruptured fuel elements and rails on other fuel elements in the tube were damaged as well. The crud film on the fuel element indicated above-normal surface temperatures over approximately 50 per cent of the surface area adjacent to the rails. It is believed at present that the primary cause of these ruptures is failure of the rails to provide an adequate bottom-side cooling annulus. The cause(s) for the observed rail damage are being investigated.

Test Program

Table VIII summarizes the hot-die-size irradiation testing that is currently in progress.

PT-IP-781-A is a run-to-rupture test which is designed to determine if the quality of hot-die-sized clad elements is comparable to the quality of AlSi elements. The test consists of 12 AlSi columns and 12 hot-die-size columns of piece-matched 0.95 w/o U-235 enriched uranium fuels in D Reactor and is designed to insure a high rejection probability if HDS fuel is twice as bad a rupture performer as AlSi fuel. The criteria for making the determination is the number and sequence of process-related failures occurring in the hot-die-sized and AlSi-clad fuel. Two ruptures have been sustained in the hot-die-size fuel in the test and none to date in the AlSi fuel. The exposures at failure were 73 per cent and 80 per cent of the goal for normal uranium. The ruptures occurred in the second and in the fifth fuel elements from the downstream end of the column. The appearance of both failures was similar,

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TABLE VIIHOT-DIE-SIZE FUEL RUPTURES

<u>Fuel Element Type</u>	<u>Reactor</u>	<u>Per Cent of Goal Exposure</u>	<u>Rupture Classification</u>	<u>Rupture Date</u>
Natural Overbore	C	25	Internal Weld Defect	8/14/65
Natural Overbore	C	69	Side Ledge Corrosion	8/21/65
Enriched Non-Bumpered	D	26	Side Groove Corrosion	8/29/65
Enriched Non-Bumpered	D	32	Side Groove Corrosion	9/ 7/65
Natural Overbore	C	71	Side Ledge Corrosion	9/11/65

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TABLE VIII

SUMMARY OF IRRADIATION TESTS IN PROGRESS
HOT DIE SIZING

<u>Test</u>	<u>Charged</u>	<u>Scheduled Discharge</u>	<u>Purpose of Test</u>
PT-IP-781-A Run-to-Rupture 408 HDS-408 AlSi Fuels 0.947 Enriched D Reactor	7-19-65	~12-15-65	Compare irradiation performance of AlSi and HDS process fuel under similar reactor conditions.
Core Optimization Test ~ 1600 HDS Elements Natural Uranium C Reactor	~10-16-65 & 11-16-65	~12-15-65 2-15-66	Evaluate behavior of HDS cores with varying amounts of Fe, Si, and Al (up to 300 ppm each) on five heat treatments: solid salt-water, hollow salt-water, hollow salt-salt, hollow salt-oil, hollow induction-oil.
High Silicon Cladding ~ 200 HDS Elements 1.25 Enriched	~12-15-65		To compare corrosion rates of X-8001 and high silicon (~10%) alloy cladding.
Heat Treatment Test 320 HDS Elements Natural Uranium	~1-15-66	~4-15-66	Compare behavior of hollow and solid induction-oil and salt-oil heat treatments.

characterized by an approximately 1-1/2 inch long by approximately one-inch wide elliptical hole in the lateral surface of the jacket, with deep corrosion of the uranium, and occurring on the "top side" of the fuel element as it rested in a horizontal position in the reactor. The ruptured fuel elements and other fuel elements in the downstream third of both tubes exhibited severe localized corrosion, known as "groove pitting" or "ledge-and-groove pitting." Penetrations estimated at approaching 20 mils deep were observed. No localized gross overheating of the surface, such as might result from fuel element misalignment, was evident. The localized corrosion was unexpected since it had not been observed in preceding irradiations of approximately 2,000 hot-die-sized fuel elements, some to over twice the exposure of the ruptured fuel elements. The phenomena of "ledge-and-groove corrosion" has occurred in and vanished from the Hanford reactors from time to time and is not well understood. A single charge of AlSi-canned controls from the test has been examined as well as other AlSi fuels from a similar environment; these exhibited only moderate or no "ledge-and-groove" corrosion. Investigation of the problem at present includes examination of hot-die-size fuels irradiated in the K Reactors as well as investigation of differences between the hot-die-size and AlSi jacketing.

The core optimization test is designed to determine the optimum alloy and heat treatment for hot-die-size cores. The test will involve elements with 13 alloys containing iron, silicon, and aluminum at 50,175 and 300 ppm and five heat treatments: hollow salt-oil, hollow induction-oil, hollow salt-water, solid salt-water, and hollow salt-salt. The cores for the test have been received at Hanford. UT-2 grain size tester data obtained by National Lead Company of Ohio has been analyzed and the predicted UT-2 peak voltage for each combination of iron, silicon, and iron are shown in Table IX. Figure 14 shows the relationship between UT-2 peak voltage and grain size. The salt-salt heat-treated cores were quenched at 932 F, and the grain size is too large to permit irradiation of this heat treatment.

The high silicon cladding test is designed to determine if aluminum cladding containing ten per cent silicon and one per cent nickel had better corrosion resistance than X-8001.

The heat treatment test is to determine the behavior of solid oil-quenched cores. All of the Hanford oil-quench experience has been with hollow heat treatments, and the economics at Fernald favor solid core blank heat treating.

REF ID: A66111

TABLE IX

EFFECT OF CHEMISTRY AND HEAT TREATMENT
ON UT-2 PEAK VOLTAGE

COMPOSITION (ppm)			HEAT TREATMENT				
			Solid Salt- Water (Volts)	Hollow Salt- Salt (Volts)	Hollow Salt- Oil (Volts)	Hollow Salt- Water (Volts)	Hollow Induction- Oil (Volts)
Al	Fe	Si					
50	50	50	1.94	.84	1.92	1.62	2.93
50	50	175	3.12	.95	2.64	3.20	3.85
50	50	300	3.76	.90	2.90	4.12	4.45
50	175	50	3.76	1.18	3.00	3.51	3.84
50	175	175	4.76	1.30	3.73	4.78	4.65
50	175	300	5.22	1.25	4.01	5.40	5.13
50	300	50	4.63	1.40	3.50	4.73	4.47
50	300	175	5.45	1.52	4.24	5.70	5.16
50	300	300	5.73	1.49	4.53	6.02	5.53
175	50	50	2.44	.89	2.25	2.37	3.32
175	50	175	3.50	.99	2.94	3.80	4.18
175	50	300	4.01	.92	3.18	4.59	4.71
175	175	50	4.11	1.25	3.31	4.00	4.14
175	175	175	4.99	1.35	4.02	5.13	4.88
175	175	300	5.32	1.30	4.27	5.62	5.30
175	300	50	4.83	1.49	3.79	4.97	4.68
175	300	175	5.52	1.60	4.52	5.80	5.31
175	300	300	5.68	1.55	4.79	5.99	5.61
300	50	50	3.07	.85	2.63	3.25	3.72
300	50	175	4.00	.93	3.30	4.54	4.51
300	50	300	4.39	.85	3.51	5.20	4.98
300	175	50	4.59	1.23	3.67	4.63	4.45
300	175	175	5.34	1.31	4.36	5.62	5.13
300	175	300	5.55	1.24	4.59	5.97	5.48
300	300	50	5.15	1.49	4.14	5.35	4.90
300	300	175	5.72	1.59	4.84	6.04	5.47
300	300	300	5.75	1.52	5.09	6.08	5.70

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FIGURE 14

RELATIONSHIP BETWEEN FEDC
GRAIN SIZE AND UT-2 PEAK VOLTAGE

BM-5300-074(3-60)
REC-66 BUCKLAND, WASH.

RL-REA-2526

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